NUCLEAR WASTES UTILIZATION

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ABSTRACT

This work presents the development of an analytical model which utilizes the thermal energy of solidified high-level nuclear waste to generate large quantities of low pressure steam while providing five-year interim storage for the nuclear waste. The fuel management, thermal-hydraulics, shielding and safety requirements for the model are investigated.

The Nuclear Waste Boiler (NWB) developed in this work was used to provide air conditioning for a large office building. The system generates steam at 30 psia to provide 2,500 tons of air conditioning, which is sufficient to cool approximately 750,000 square feet of floor space.

The design model was compared economically against a fossil fueled boiler of equivalent capacity. The analysis indicated that considerable savings can be realized by using the NWB as a heat source. The savings depend upon many variables, but the most significant parameter is the fossil fuel cost. Specifically, if fossil fuel costs increase 15% annually, the NWB will represent a $255,000 annual savings at 10% over a 15-year economic life.

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I. INTRODUCTION

A. Background

As the nuclear power industry assumes a more important role in the nation's electrical energy requirements, the quantity of radioactive waste will increase by several orders of magnitude. The installed nuclear capacity in the United States is expected to increase from 134 gigawatts in 1980, to 504 gigawatts in 1990 and to 1200 gigawatts in the year 2000. It is estimated that the equivalent of 471,000 ft^3 of high-level solidified wastes will have been generated by the year 2000[1]. These high-level wastes consist principally of the fission-product concentrates that arise in the reprocessing of spent reactor fuels, and are characterized by their intense, penetrating radiation and relative high thermal power.

The policy of the United States Government is that the Atomic Energy Commission (AEC) will take custody of all commercial high-level nuclear waste and dispose of them in perpetuity[2]. In implementing this waste management policy, the AEC has published an Appendix to reference 3 which requires industry to solidify their high-level liquid wastes and ship them to the Federal Repository within ten years after reprocessing. The projections of the accumulated volume of high-level solidified waste[1] are shown in Figure 1-1.

Both the burial of high-level wastes in natural salt formations and the Retrievable Surface Storage Facility (RSSF) are under investigation as final disposal concepts, but the ultimate disposal sites or methods have not been selected. Regardless of method, it is generally

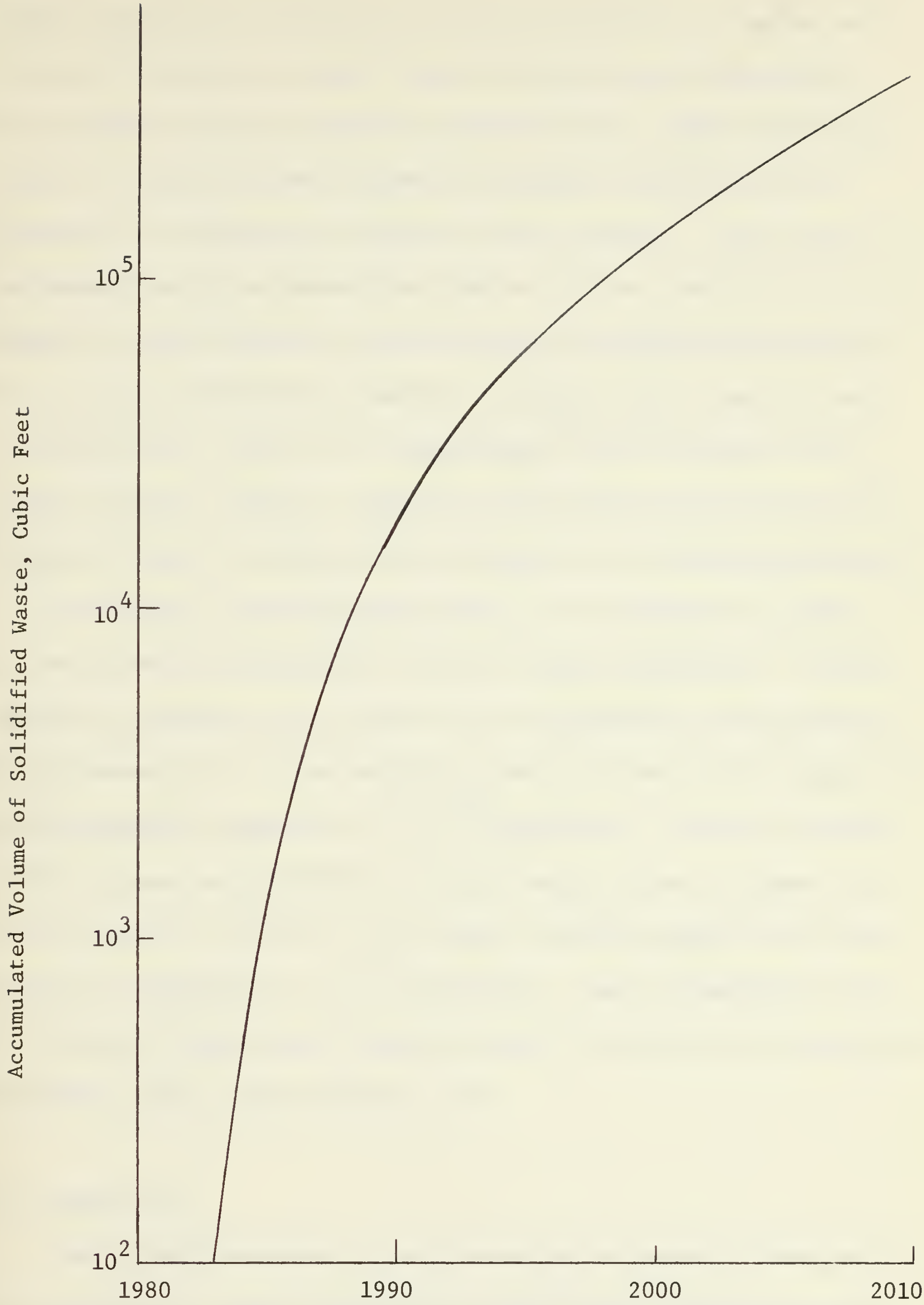


Figure 1-1

Projection of the Accumulated Volume of High-Level Solid Waste at Federal Repository after Ten Years Interim Storage

agreed that the waste should be stored in solid form. Storage as a solid provides greater safety because the solids are less mobile, less soluble in water, smaller in volume and more rugged physically. Figure 1-2 shows the heat content of a typical solidified waste cannister as a function of time out of the reactor[4]. This figure is representative of the waste from light water power reactor fuel irradiated to 45,000 MWD/MT at a power level of 30 MW/MT and solidified by the spray solidification method. The figure shows that the heat rate decays with time, but is significantly high immediately after removal from the reactor. Because of this high initial heat content, interim storage is required before delivery to the Federal Repository.

During this interim storage period, it is necessary to remove the heat from the cannisters of waste to keep them intact. The heat removed is generally not utilized but is dispersed to the atmosphere. From reference 5 it is estimated that the total cost of the high-level waste management program is 39×10^{-3} mill/kwhr. This cost includes liquid storage, solidification, interim solid storage, shipment, and final disposal. The cost of five years interim storage of the solid waste was estimated at 3×10^{-3} mill/kwhr. Thus utilization of the heat produced during solid interim storage will reduce the total waste management cost by approximately 7.7%.

B. Objectives

The primary objectives of this work are to develop a model which will utilize the thermal energy of the high-level radioactive waste during interim storage to generage low pressure steam, and to economically compare this model with an equivalent fossil-fueled

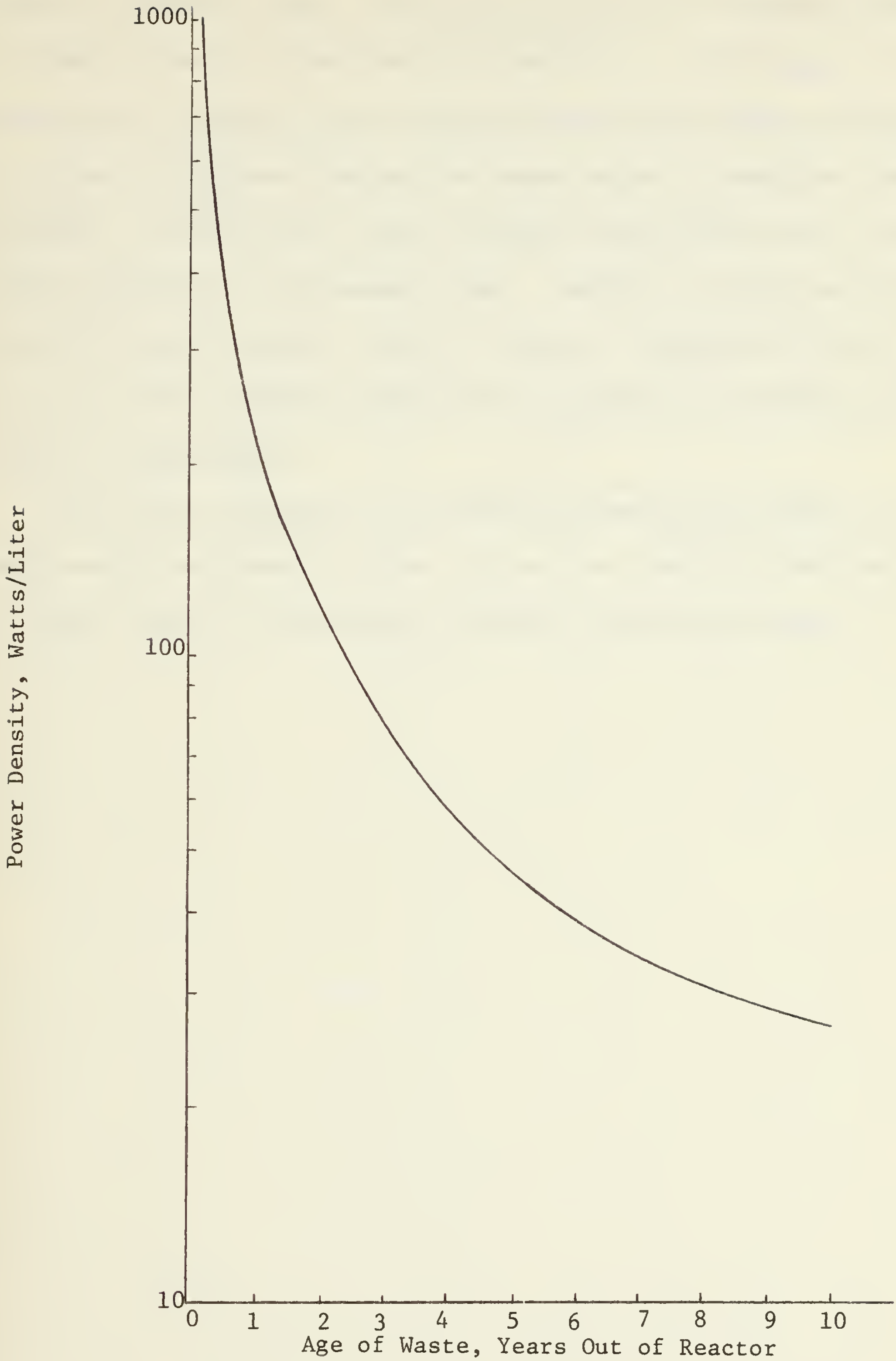


Figure 1-2

Power Density of Spray Melt Solidified High-Level Waste

system in providing steam. The model design will be compatible with the RSSF Concept B as described in reference 6. In this work, the model will be referred to as a Nuclear Waste Boiler (NWB), and will be fueled by high-level radioactive wastes sealed in stainless steel cylinders. The NWB can be used in numerous applications which require large quantities of low pressure steam. However, in this application it will be used to provide steam at 30 psia to absorption chillers and to hot water converters which will be capable of providing year-round air conditioning.

The plant is composed of the NWB and steam supply system, the fuel storage and transfer system, and the heat removal safety system. The plant, itself, is described in detail in the next section.

II. PLANT DESCRIPTION

Plan and cross sectional elevation views of the NWB plant layout are shown in Figures 2-1 and 2-2. It is seen in Figure 2-1 that the plant is divided into four general areas. The four areas are labeled at the top of Figure 2-1. The first area is the receiving area where the fuel elements in their shipping casks are received from the fuel reprocessor. The second division is the fuel storage pool which is further divided into two sections. Section B of the fuel storage pool is the normal fuel storage area. It will contain all spent fuel removed from the boiler as well as all new fuel received. An isolation cell used to check the fuel elements for leaks is also contained in this section. No fuel will normally be stored in Section A of the storage pool. However, this section is capable of storing all the fuel elements in the boiler in an emergency. This section primarily functions as a heat sink for the heat removal system.

The NWB itself comprises the third division of the plant. The NWB is composed of four complete boiler sections. Each section is identically designed to produce steam independent of the remaining sections. Each section will contain a primary and an emergency heat removal system. The primary heat removal system must be manually activated and will normally be used when refueling or when the boiler section is down for an extended period. The emergency heat removal system is automatically actuated by motorized valves when a loss of flow or when a fission product release is sensed by the control system.

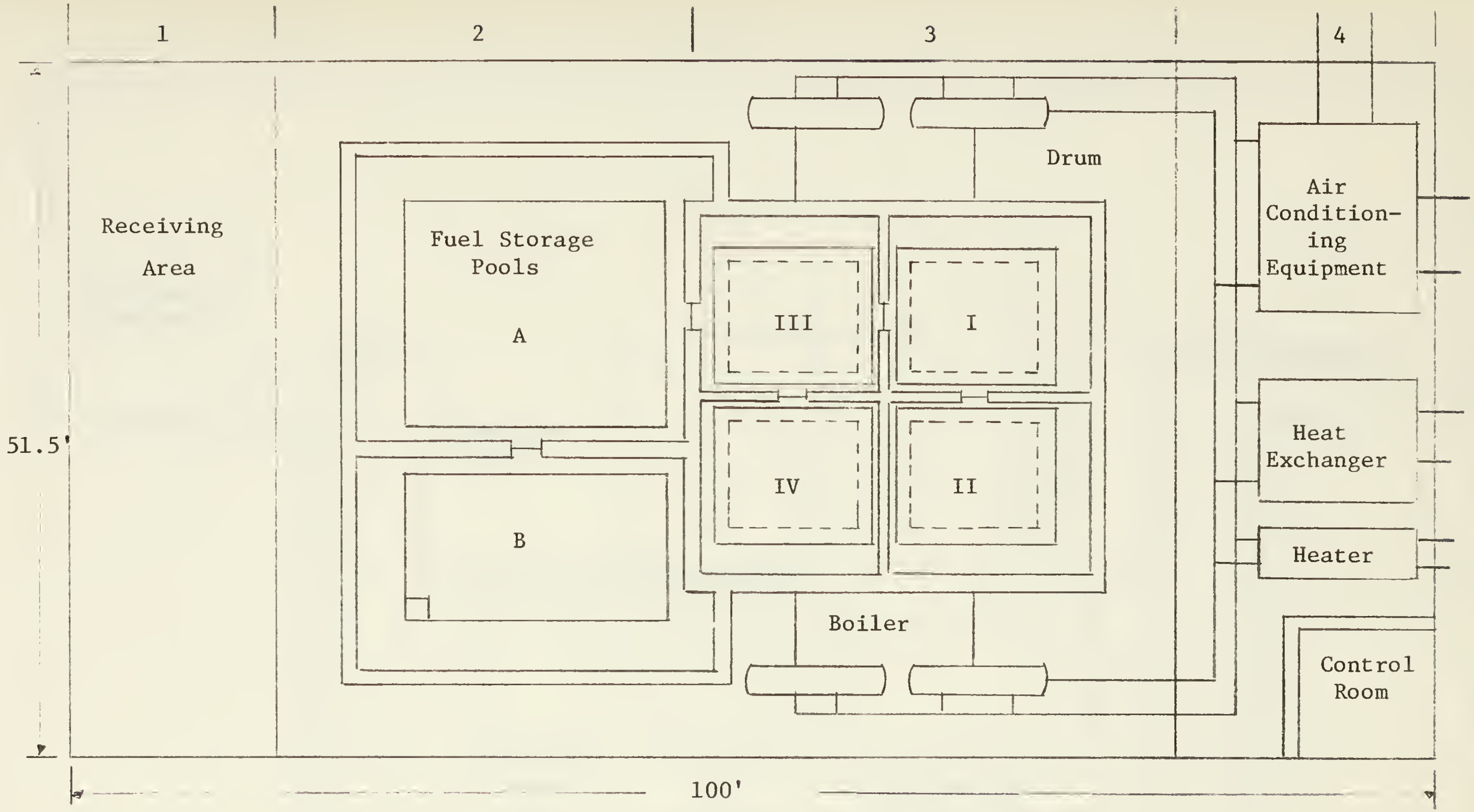


Figure 2-1

Plan View of NWB Plant

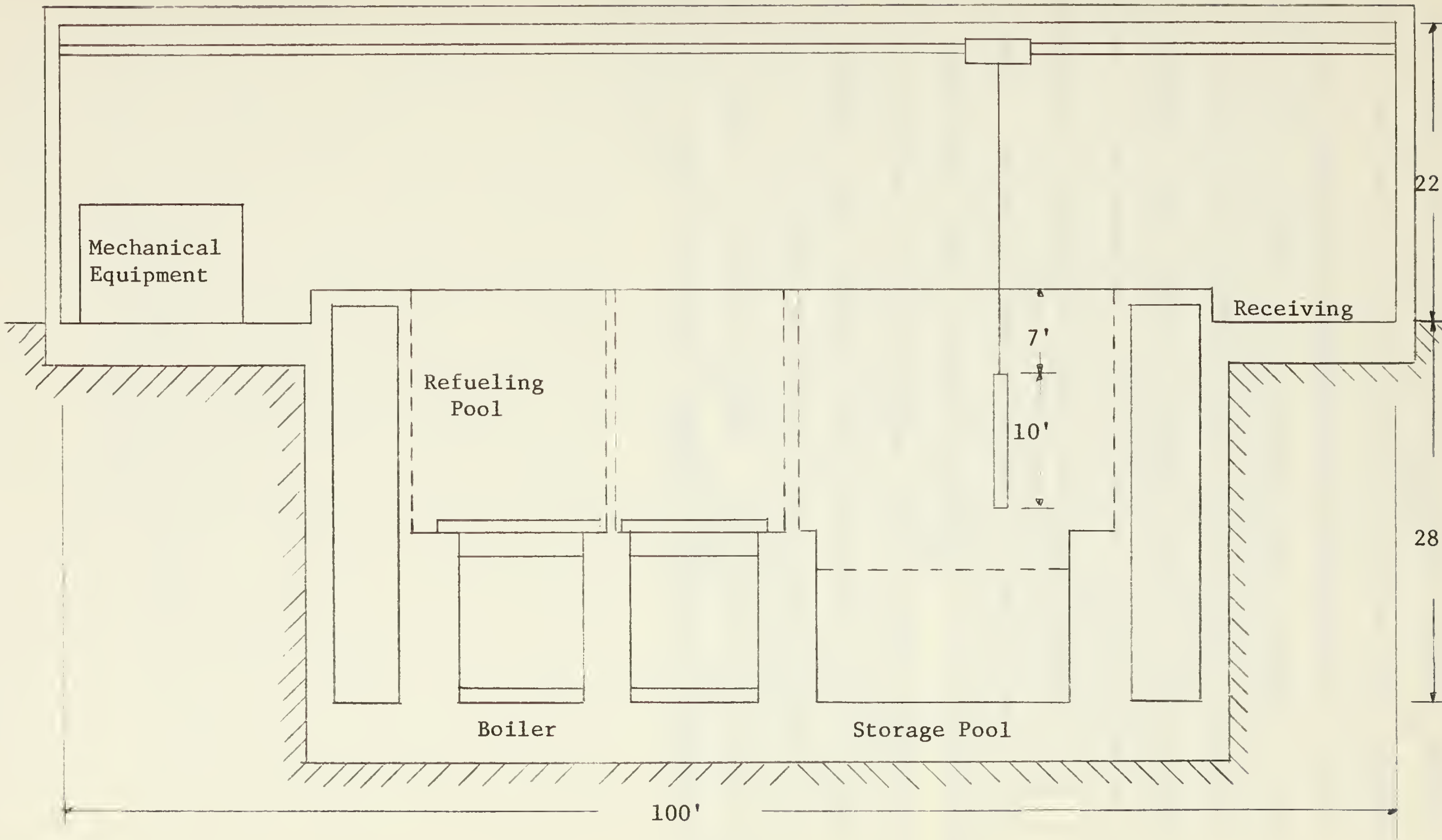


Figure 2-2

Elevation View of NWB Plant

The final plant division is the mechanical equipment section. This area contains the absorption chillers, the hot water converters and the heat removal heat exchangers and cooling towers. The air conditioning is provided by the absorption cycle chillers. In these units the steam from the NWB is used to boil the water refrigerant. The refrigerant vapor which was separated from the lithium bromide absorber is then condensed. The refrigerant is again evaporated in the presence of the absorber which has a great affinity for the refrigerant. During the evaporation heat is removed from the chill water system which passes through the evaporator. This chilled water is then circulated to provide cooling. The hot water converter is a heat exchanger which utilizes the NWB steam to heat the circulating hot water. When neither the chilled or hot water is required, the steam is dumped in the heat exchanger. Only the hot and chilled water from the AC system and the heat exchanger coolant are permitted to leave the plant.

III. FUEL MANAGEMENT

A. Fuel Element Design

The fuel used in the NWB will be solidified high-level nuclear waste hermetically sealed in stainless steel cylinders. In the United States considerable work has been conducted in the development of four processes for solidifying high-level wastes. These processes are pot calcination, spray melt solidification, phosphate glass solidification and fluidized bed calcination. Each method has been developed to the point of demonstration on an engineering scale.

A summary of the characteristics of the solids produced by each of the above processes is shown in Table 3-1[2]. To insure that the solid waste is not permitted to enter the human environment, certain characteristics in the solid waste are desirable. The characteristics of primary importance are high stable temperature, high thermal conductivity, low leachability in water, good chemical stability and radiation resistance, mechanical ruggedness, and noncorrosiveness to container. Minimum volume and minimum production costs are economically important in selecting the solid product for fuel elements.

A review of Table 3-1 shows that the spray melt solidification process provides a solid product with the greatest number of desirable safety and economic characteristics. This product has a greater thermal conductivity (.4 to 1 BTU/hr-ft-°F), a higher maximum stable temperature (900°C) and is hard and tough with the low leachability in cold water of 10^{-3} to 10^{-6} gm/cm^2-day.

Table 3-1

Properties of Solidified High-Level Waste[2]

	Pot Calcine	Spray Melt	Phosphate Glass	Fluidized Bed
Form	Calcine Cake	Monolithic	Monolithic	Granular
Description	Scale	Microcrystal	Glass	Amorphos
Thermal Conductivity Btu/hr-ft-°F	0.15 to .25	.4 to 1.0	.4 to 1.0	.1 to .25
Bulk Density, gm/ml	1.1 to 1.5	2.7 to 3.3	2.7 to 3.3	1.0 to 1.7
Maximum Heat Rate, Watts/Liter (during test run)	85	204	190	70
Leachability in Cold Water gm/cm^2-day	1.0 to 10^{-1}	10^{-3} to 10^{-6}	10^{-4} to 10^{-7}	1 to 10^{-1}
Hardness	Soft	Hard	Very Hard	Moderate
Friability	Crumbly	Tough	Brittle	Moderate
Maximum Stable Temp., C	900	900	500	600
Container Material	Stainless Steel	Stainless or Mild Steel	Stainless or Mild Steel	Stainless or Mild Steel

The spray melt solidification method was developed at Battelle-Northwest[5]. In this process the liquid waste, containing some or all of the melt-forming additives, is fed through a pneumatic atomizing nozzle into the top of a heated cylindrical tower. The atomized waste is evaporated, dried and calcined to a powder as it falls into a melter where it is melted at temperatures up to 1200°C. A monolithic solid is formed after the melt is cooled. The solid is a tough micro-crystalline material having a good thermal conductivity and a moderately low solubility in aqueous solution.

A typical cannister of the solidified waste used as fuel elements in the NWB will be 8.65 inches in diameter and 10 feet in length. The power density of the waste cannister as a function of time out of the reactor was shown in Figure 1-2. A summary of the fuel elements properties is given in Table 3-2 below.

Table 3-2

Properties of NWB Fuel and Container Materials

Fuel Material:	
Thermal conductivity, BTU/HR-PT-°F	0.646
Bulk density, lb/ft^3	206
Melting point, °F	1652
Volume of fuel per element, ft^3	3.49
Distribution in containers	Uniform
Containers:	
Material:	Type 304 Stainless Steel
Inside diameter, inches	8.0
Outside diameter, inches	8.65
Wall thickness, inches	0.325
Active length, feet	10

B. Fuel Cycle

The NWB fuel cycle begins with the production of the fuel elements at the reprocessing plant and ends with the shipment to permanent storage of the elements removed from the boiler. Two important factors must be considered when determining the fuel cycle. First, the age of the fuel at startup and the length of time in the boiler must result in the maximum possible steam production. Secondly, the length of time in the boiler must be compatible with the proposed interim storage time. Figure 1-2 shows that the power density in the waste product initially decreases rapidly. The power density decreases from 220 w/l at the end of one year, to 130 w/l at two years, and to 47 w/l at the end of five years. Beyond the five year point, the power density is too low to economically produce steam. Therefore, it is important to place the fuel into service as soon after reprocessing as possible.

The initial heat generation rate in a solidified waste may not be compatible with economic heat dissipation in a final storage environment; consequently, an aging or interim storage period may be required. The NWB would provide safe interim storage of the solidified waste and would decrease the cost of final storage. By aging the waste in the NWB the containers can be loaded with a higher heat generation rate than the final storage environment may allow. The waste would be aged in the boiler until it reaches a heat generation rate compatible with the final storage requirements. Several studies have indicated that five to ten years interim storage of the solid waste is desirable[2,5].

Therefore, to be beneficial both as a heat source in steam generation and as an interim storage facility, a five year fuel cycle would be required. This cycle includes six months for the accumulation of the fuel elements to charge a boiler section, four years lifetime in the boiler, and six months storage awaiting shipment to permanent storage.

To determine the power density of the fuel during its lifetime in the boiler, several assumptions and simplifications were used. It was assumed that the reactor fuel was reprocessed immediately after removal from the reactor and shipped to the boiler sites. The fuel elements are accumulated, placed in the boiler and begin their boiler lifetime at seven months after removal from the power reactor. All calculations were performed on a monthly interval using the mid-interval power density as the average power density. The power density curve in Figure 1-2 was estimated by a least square polynomial fit, and the power densities used ranged from 319.5 w/l for the first month to 50.6 w/l for the last month in the NWB.

C. Annual Steam Requirement

As stated earlier the proposed application of the NWB in this work is to provide steam for air conditioning requirements for a large office building. A detail mechanical design was not performed in this work. Instead the actual steam requirements for a similar but smaller system were used to determine the steam requirements. The actual system was an office building which provided 60,000 ft^2 of floor space and required 186 tons of air conditioning[7]. The steam

usage for the building was determined assuming a maximum steam demand of 1.577×10^6 lb/month. The normalized steam curve was developed from this and is shown in Figure 3-1. The design system is 2500 tons of air conditioning. Adjusting the steam demand curve to design data, the peak value for the design steam curve becomes 2.12×10^7 lb/month.

Based on information obtained from absorption chiller manufacturers an absorption air conditioning unit requires 18 lb of steam per hour per ton of refrigeration. Assuming an 80% load factor, the peak demand is 36,000 lb/hr. This value was used to determine the fuel requirements for the NWB. The average fuel element power density at beginning of equilibrium cycle is 156 w/l or 1.5×10^4 BTU/hr-ft^3. The heat required to produce 36,000 lb of steam per hour is 3.4×10^7 BTU/hr. Equating the above

$$Q = q''' V$$

or

$$V = \frac{Q}{q'''} = \frac{3.4 \times 10^7}{1.5 \times 10^4} = 2200 \text{ ft}^3$$

Since there are 3.49 ft^3 per fuel elements, approximately 650 fuel elements are required. For symmetry an array of 13 by 13 fuel elements in each boiler section or a total of 676 fuel elements will be used.

The steam production calculations were formed for the boiler loaded with 676 fuel elements. The results show that the maximum steam produced is 37,453 lb/hr which exceeds the maximum steam demand. The peak month steam production is 26,966,400 lb. This also exceeds

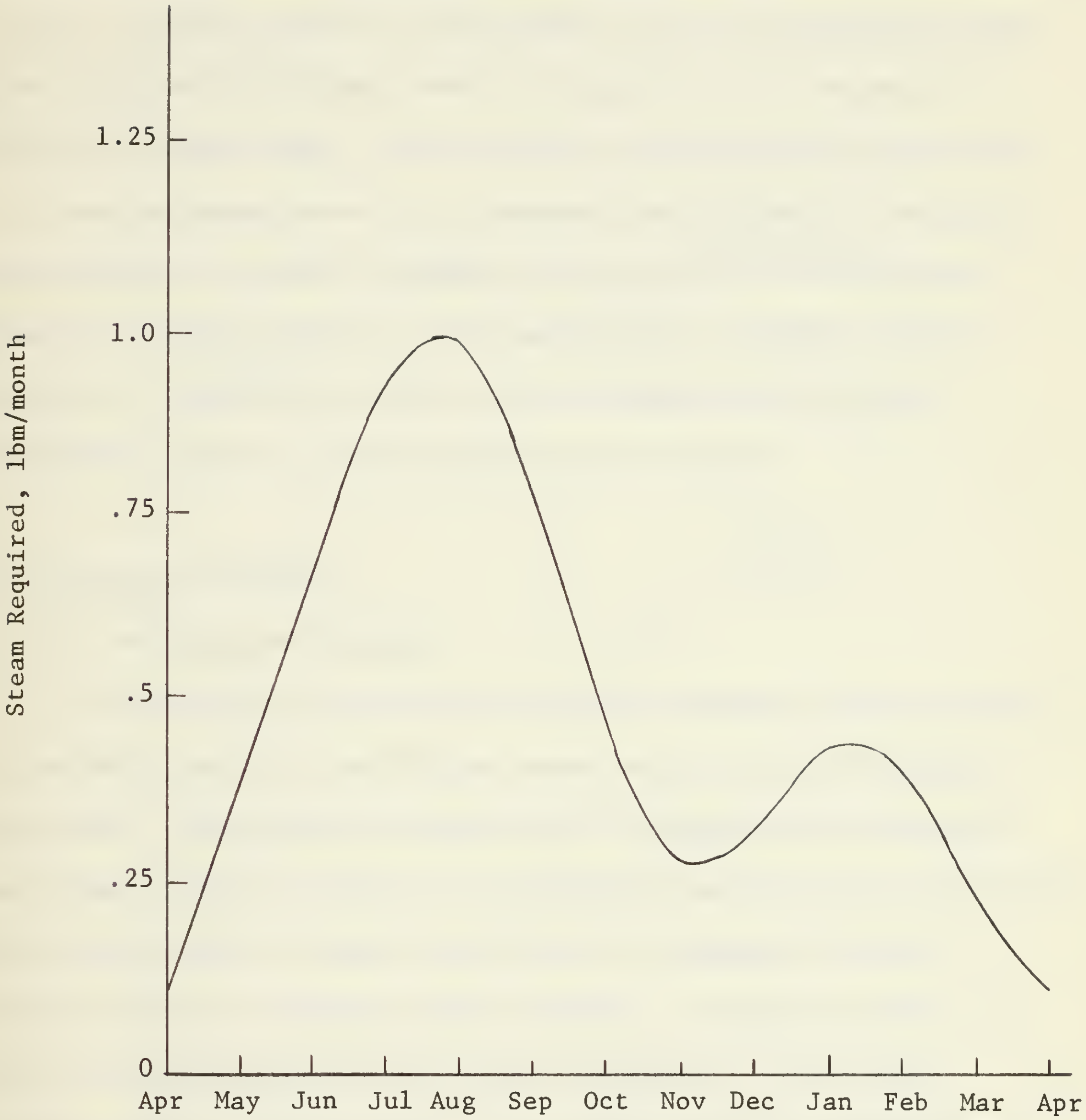


Figure 3-1

Normalized Steam Demand
Peak Value is 2.12×10^7 lb/month

the maximum steam demand. The steam production data was normalized and is shown in Figure 3-2, superimposed on the steam demand curve. It shows that the steam produced always exceeds the steam demand, even when the boiler is being refueled just prior to the peak demand in July. At this time 1/4 of the boiler will not be producing steam. Since the steam production cannot be regulated, it is necessary to "dump" the excess steam. This is done by diverting the excess steam to a heat exchanger where it is condensed and returned to the boiler. The addition of the heat exchanger will not add to the system cost because for safety reasons a full capacity heat exchanger will be required. However, the system's attractiveness will be greatly improved if a use for the excess steam can be found.

D. Refueling Procedures

1. Receiving and Shipping

The high-level solidified waste fuel elements will be received at the NWB plant in large shipping casks which provide radiological protection. The shipping container will be placed in the receiving pool where it will be opened. The recirculating water in the receiving pool is monitored to insure that there is no leakage of fission products from the stainless steel cylinder. If leaks do exist, the cask is resealed and returned to the reprocessor. If the fuel element passes the monitoring, it is removed from the shipping cask and placed in pool storage to await boiler loading. A spent fuel element is then placed in the shipping cask which will be sealed and removed from the receiving cell and shipped to permanent storage.

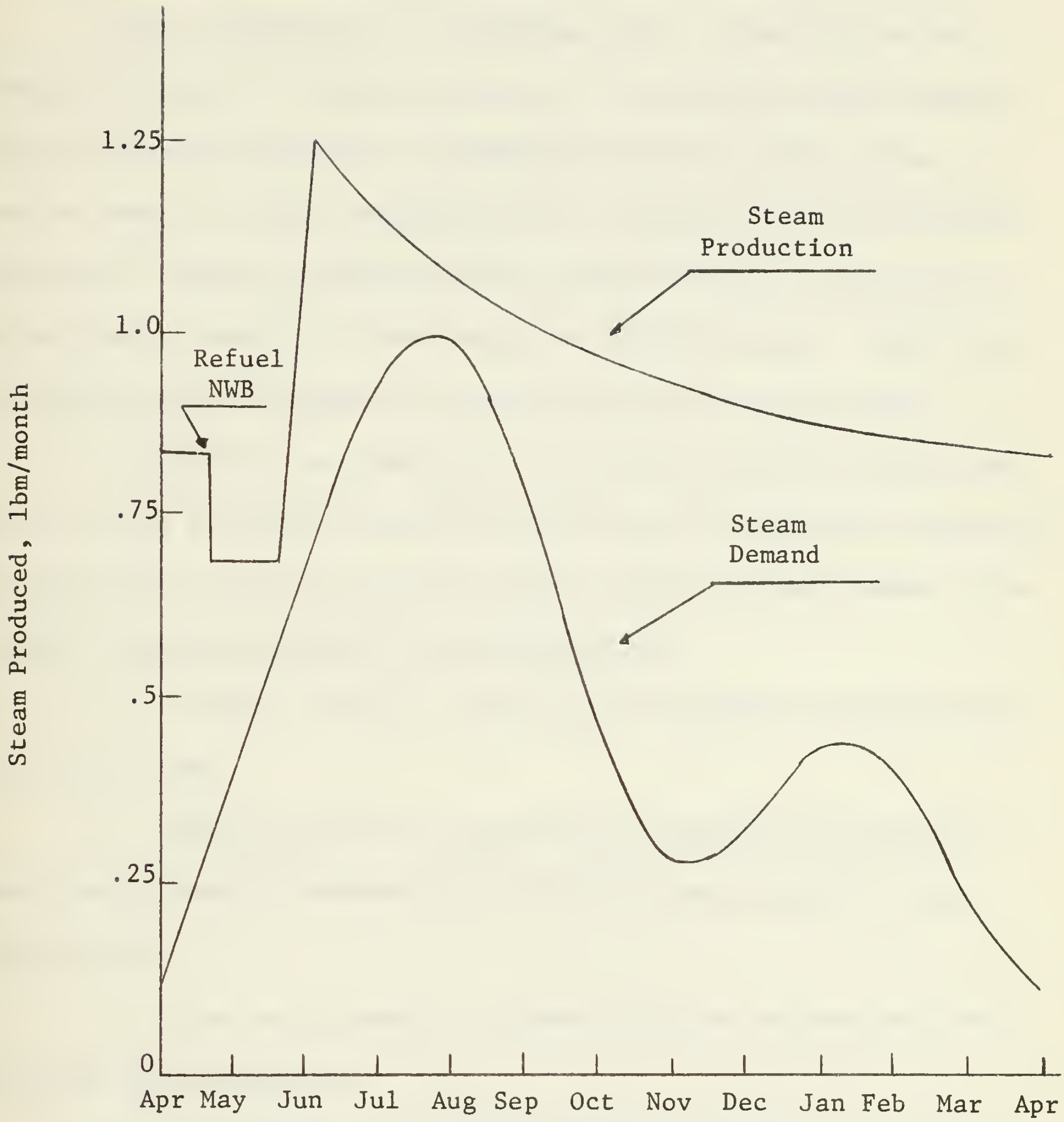


Figure 3-2

Normalized Annual Steam Production
Peak Value is 2.7×10^7 lbm/month

This loading and unloading will be performed under adequate water shielding, with the aid of a 35 ton overhead traveling bridge crane.

2. Boiler Loading

Boiler loading must be performed with a refueling machine. Greater precision is required because of the closely spaced lattice within each boiler section. An overhead traveling bridge crane of one ton capacity with a refueling tool is adequate for the refueling procedure. However, a remote control closed circuit TV system will be required to assist in positioning the refueling tool. The 35 ton crane will also be required to remove the boiler section cover.

The NWB will be scheduled for refueling annually during June. The fuel will be ordered several months before the scheduled refueling to insure that the fuel will be on hand to meet the peak demand. The general refueling procedure is outlined below:

a. The cover over the section to be refueled is removed with the 35 ton crane.

b. Using the one ton crane with the refueling attachment, a spent fuel element is removed from the boiler and placed into the storage pool.

c. A new fuel element is removed from the storage pool and is placed into the boiler.

d. This procedure is repeated until all the spent fuel has been replaced.

e. The section cover is replaced and secured, and the section is returned to operation.

It should be noted that both the boiler and the storage pool have sufficient water cover height to allow transfer of the fuel elements within acceptable radiation levels. Also, during the actual fuel transfer, personnel will be excluded from the boiler room.

Glossary

Q = Total heat, BTU/hr

V = Fuel volume, ft^3

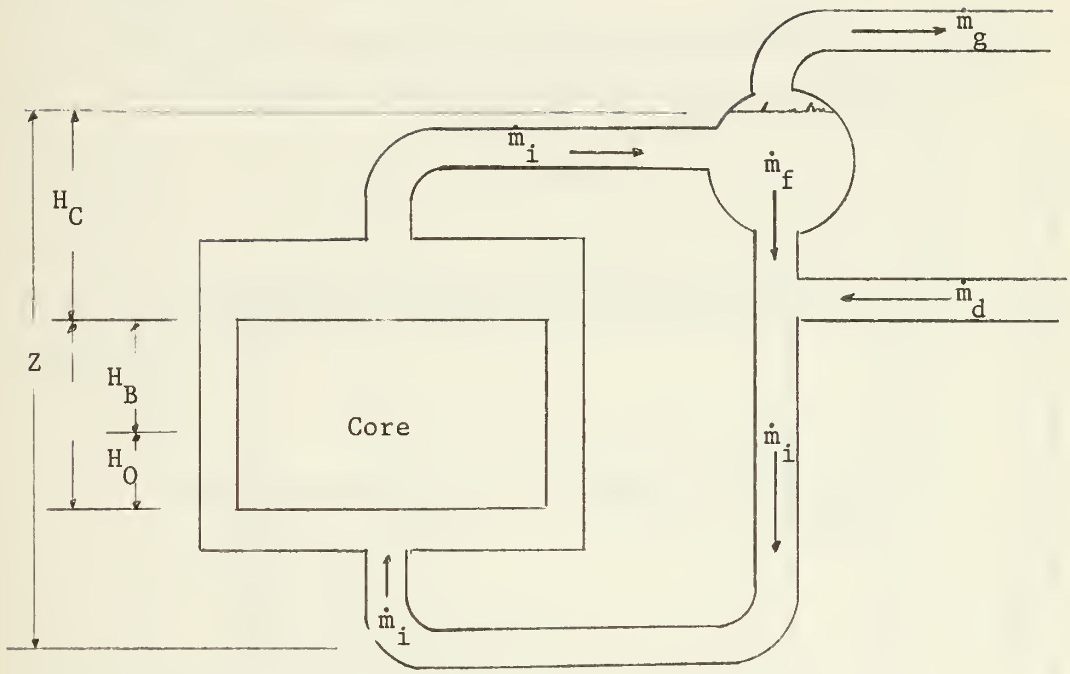
q''' = Power density of the fuel, BTU/hr-ft^3

IV. NUCLEAR WASTE BOILER DESIGN

The NWB plant was designed similar to a natural circulating boiling water reactor. The design model is shown in Figure 4-1. In this model the condensate enters the reactor loop just below the steam drum at a rate of $\dot{m}_d$ lb/hr. This flow is mixed with the recirculating flow, $\dot{m}_f$ lb/hr, and the combined flow, $\dot{m}_i$ lb/hr, enters the reactor at the bottom and flows upward through the core. The feedwater receives sensible heat in the non-boiling height of the core, and latent heat of vaporation in the boiling height of the core. The two-phase mixture of liquid and vapor leaves the core and is separated in the steam drum. The saturated water is recirculated and the steam, $\dot{m}_g$ lb/hr, is distributed, condensed, and returned as condensate. Unlike a reactor, the heat in the NWB is added linearly because of the uniform composition of the fuel.

A. Boiler Design

The NWB is designed to provide four years interim storage for the high-level nuclear waste used as fuel elements. One fourth of the fuel in the boiler will require annual replacement. To accommodate this cycle the boiler was divided into four equal sections. The sections were designed as identical modules capable of producing steam independent of the remaining sections. This will allow three boiler sections to produce steam while the remaining section is being refueled. A schematic of a boiler section is shown in Figure 4-2. Each module contains a pressure vessel, steam drum, circulating



Z	=	Reactor Height	$\dot{m}_g$	=	Steam Flow
H	=	Core Height	$\dot{m}_f$	=	Recirculating Flow
H_C	=	Chimney Height	$\dot{m}_d$	=	Condensate Flow
H_O	=	Non-Boiling Height	$\dot{m}_i$	=	Core Flow
H_B	=	Boiling Height			

Figure 4-1

Natural Circulating Reactor System

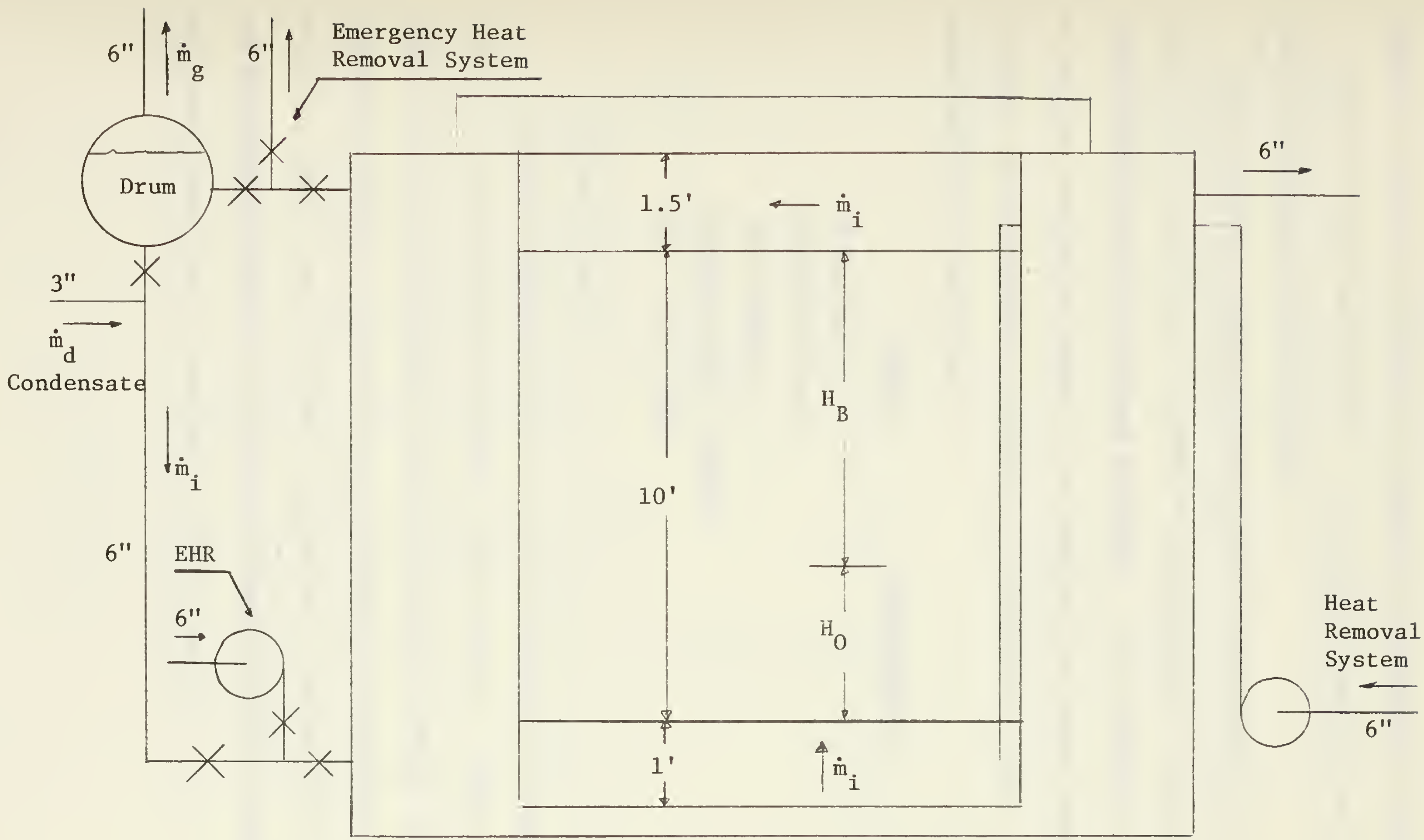


Figure 4-2

Cross Sectional Elevation of Nuclear Waste Boiler Section

loop, heat removal loop and a backup safety loop. Plan and cross sectional elevation views of the NWB circulation system are shown in Figures 4-3 and 4-4. Since all sections are identical, the systems are shown for only one section. All related dimensions and specifications are indicated on the figures.

The circulation system and the other piping systems in the NWB contain numerous valves. In order to simplify later discussions, a valve identification system was developed. All valves are identified by section, system and number. For example, the second steam valve in boiler section number one would be identified as 1-S-2. The letter codes used are listed below:

System	Letter Code
Steam Supply	S
Condensate Return	C
Heat Removal	H
Emergency Heat Removal	E
Pool Heat Removal	P

1. Pressure Vessel

The NWB pressure vessel is a cast-in-place reinforced concrete structure that is divided into four equal sections. Each section is lined with 3/16 inch stainless steel and will be covered by a post-tensioned concrete cover which can be removed for refueling. The stainless steel liner provides a high degree of water tight integrity and the massive concrete structure provides biological shielding. Additional shielding is provided by a water cover.

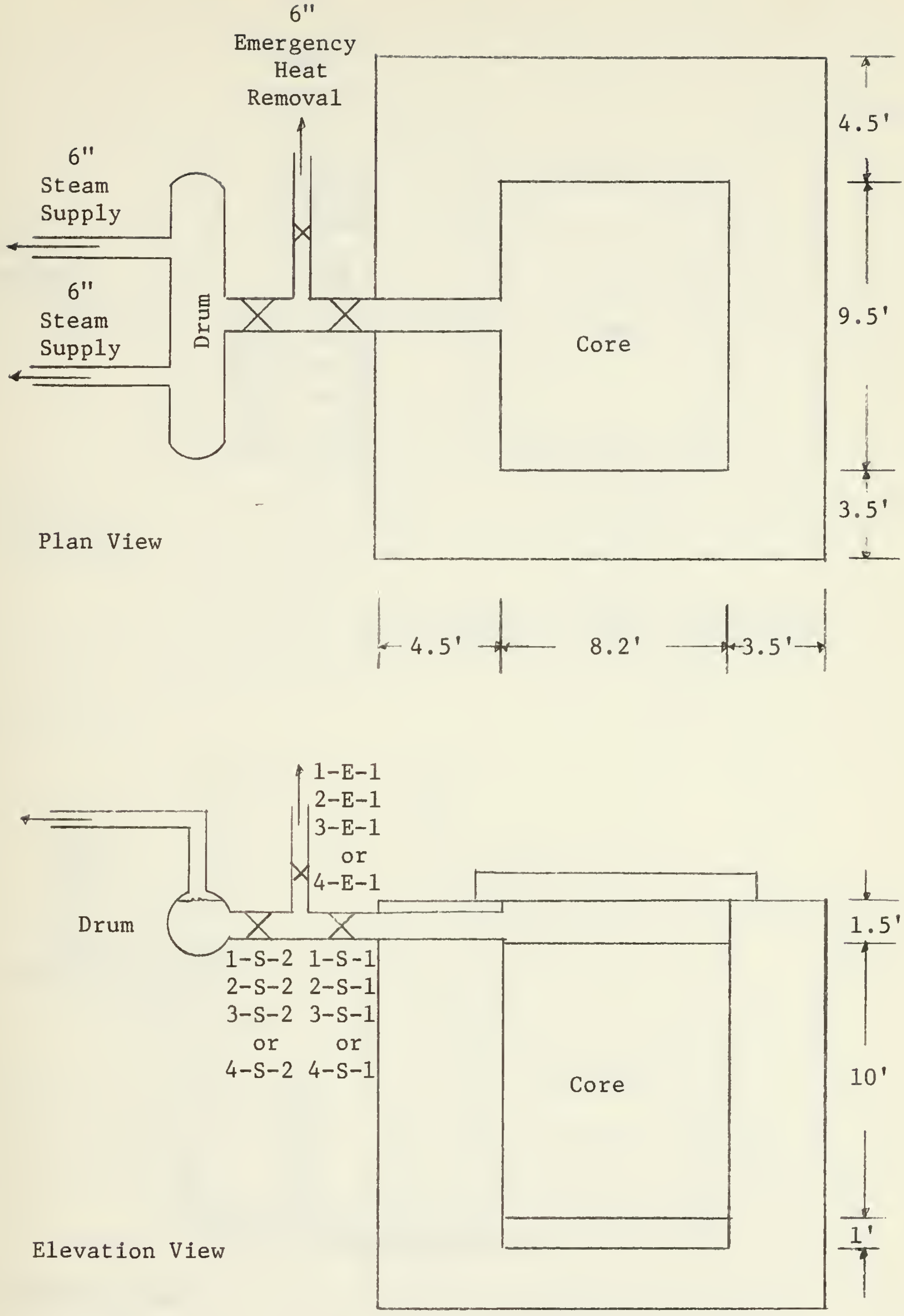


Figure 4-3

Steam Supply System

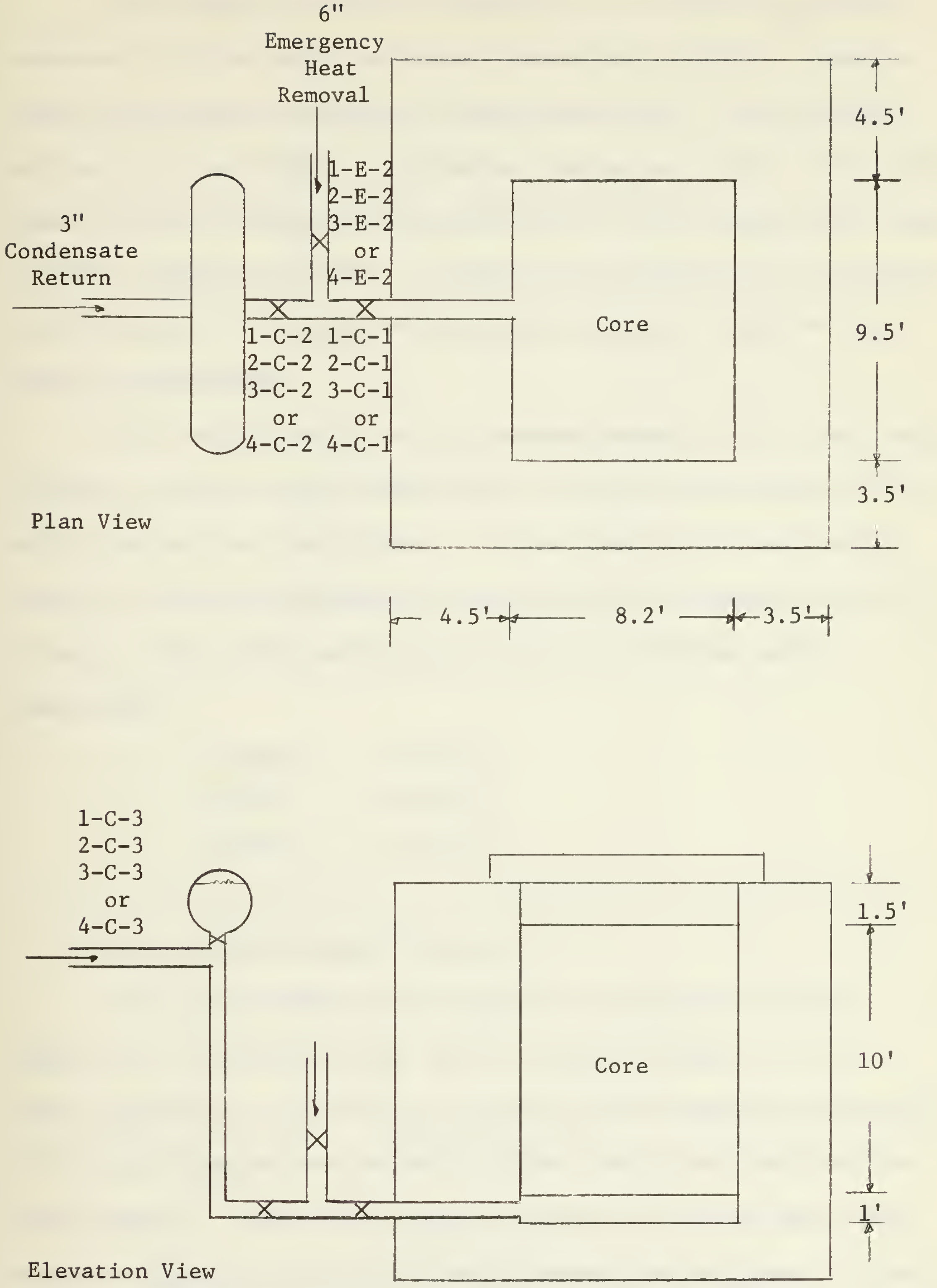


Figure 4-4

Condensate System

The pressure vessel was designed for an internal operating pressure of 30 pisa and a saturated temperature of 250°F. Pressure relief devices are installed to protect the system. For simplicity and ease of construction, a rectangular design was chosen. The pressure vessel and the refueling pool located above it are located below grade to reduce radiation hazards. To reduce the possibility of accidental core drainage, all pressure vessel penetrations are above the core, except the downcomer.

The internal height of the pressure vessel is 12.5 feet, which provides a one foot lower plenum and a 1.5 foot upper plenum at each end of the ten foot fuel element. As was shown in the fuel design section, each boiler module will contain 169 fuel elements in a 13 by 13 array. The boiler sections will have the following inside dimensions:

Length:	9.5 feet
Width:	8.2 feet
Height:	12.5 feet

2. Fuel Element Support Structure

The fuel elements are arranged in a triangular lattice, supported by a stainless steel grid as shown in Figure 4-5. The fuel cylinders fit into the base support and are stabilized laterally by one inch diameter rods extending three feet up the length of the fuel elements. Each boiler section contains 169 nuclear waste fuel elements placed in a 13 by 13 closely packed array as shown in Figure 4-6. This close packing is required because of the relatively low power density.

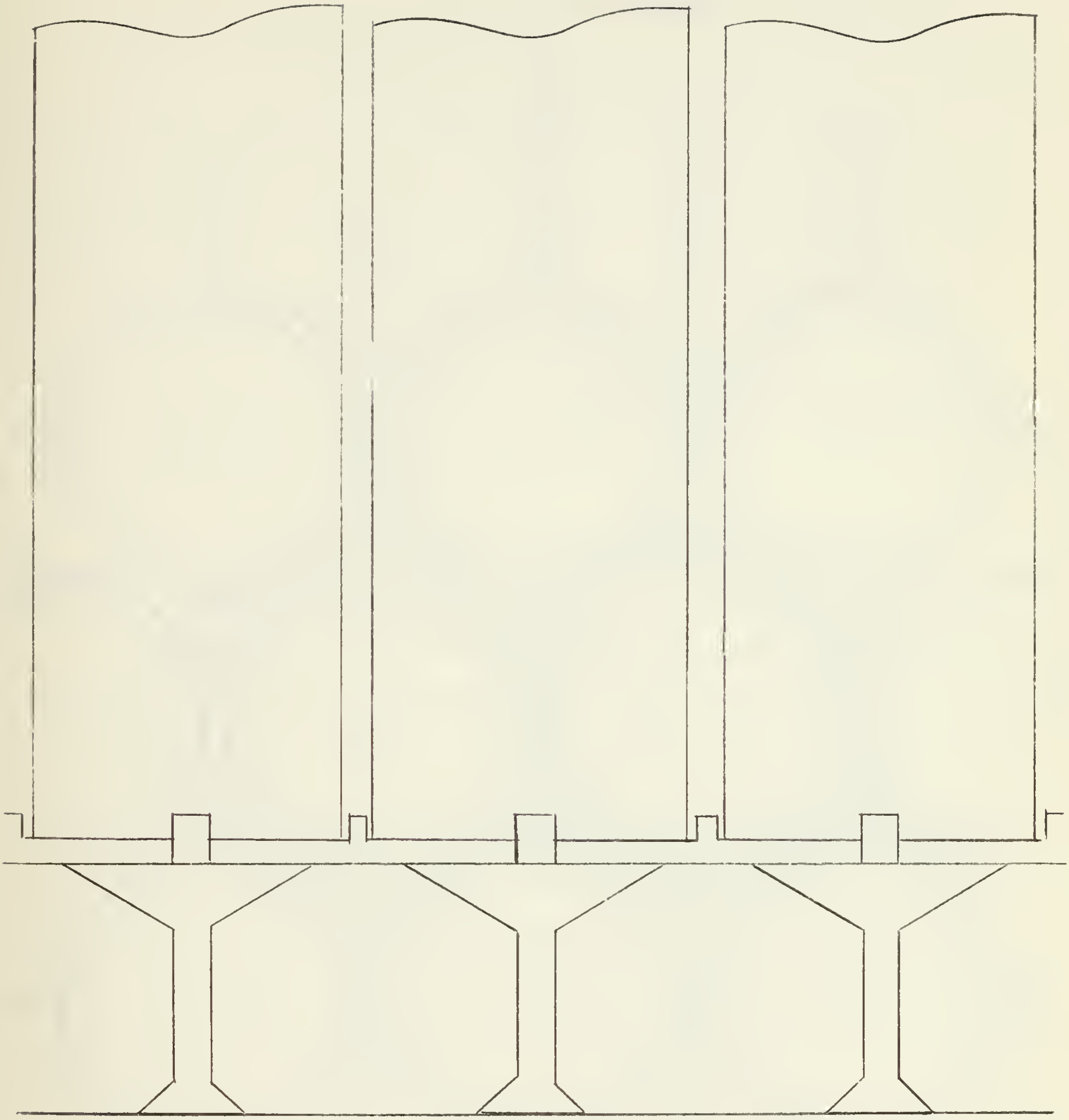


Figure 4-5

Elevation View of Fuel Element Support Structure

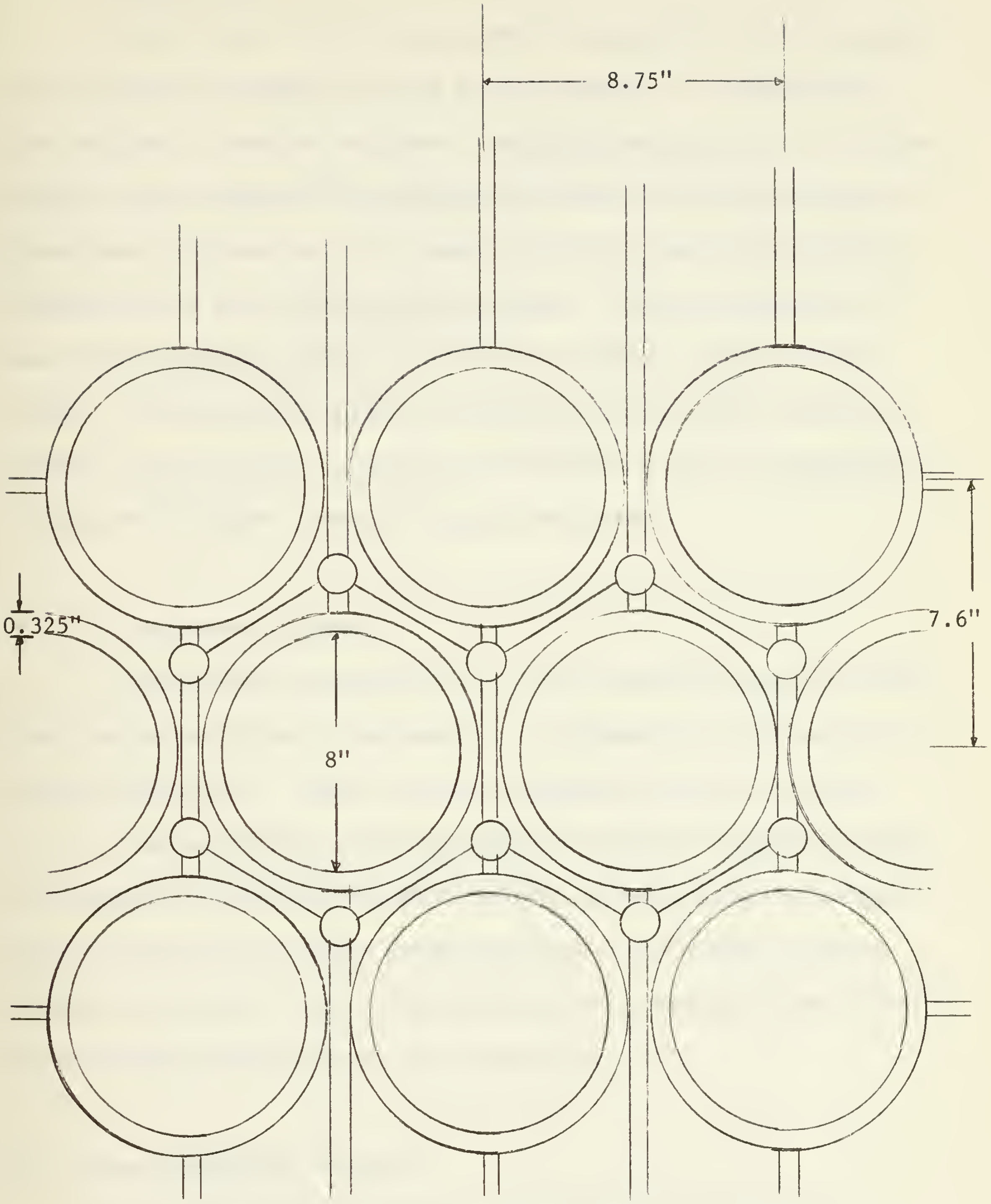


Figure 4-6
Fuel Element Lattice

3. Steam Drum

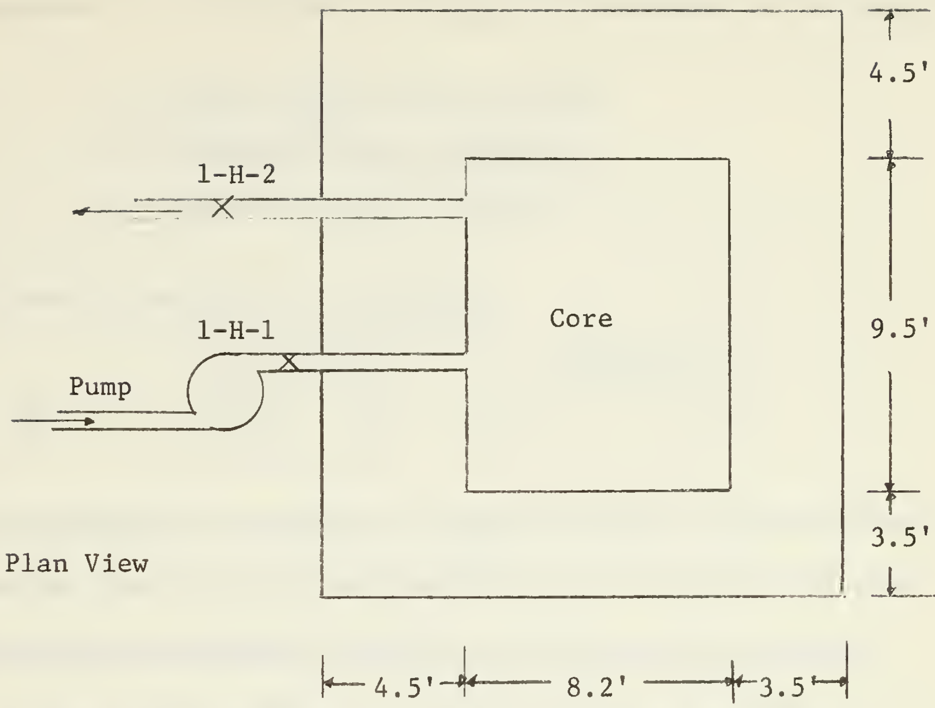
Each boiler section has a gravity separation steam drum which is 24 inches in diameter and ten feet in length. All penetrations are baffled to restrict carryover. Moisture carryover into the steam supply system reduces the equipment efficiency and the carryunder of vapor into the downcomer will reduce the fluid density in the downcomer and will reduce the driving pressure. The drum contains two six inch steam supply lines, a six inch downcomer, and a 12 inch riser. The downcomer contains a variable orifice which is used to control the boiler flow. The 12 inch diameter riser was required to reduce the two-phase pressure loss in the system.

4. Heat Removal System

Each module is equipped with a heat removal system to remove heat during refueling or emergencies. A schematic of the system is shown in Figure 4-7. Under normal operations the heat is removed by the steam generation and condensation. However, during refueling or emergency conditions, the heat removal system can be activated to keep the fuel in the proper operating range. The system uses the storage pool water, which is always kept cool through routine circulation through cooling towers, as a heat sink.

B. Thermal-Hydraulic Analysis

A thermal-hydraulic analysis was performed on the natural circulating steam generating system. A computer code developed in reference 8 was used in the analysis. This code calculates the



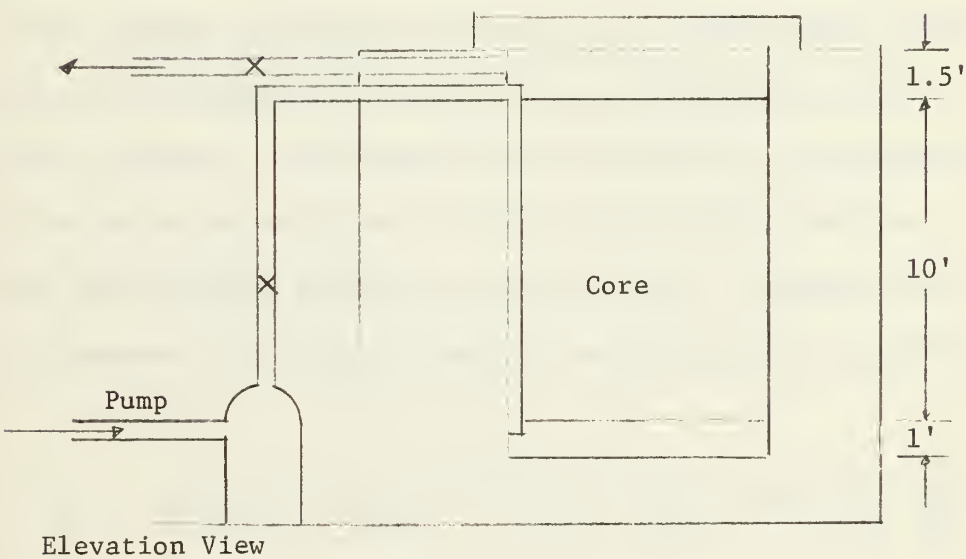


Figure 4-7

Heat Removal System

driving pressures, system pressure losses and flow rates based on the given parameters. The following assumptions were made in the analysis:

1. System pressure was 30 psia.
2. Saturated inlet conditions.
3. Constant 5% exit quality.

The code uses the emperical relationship

$$\frac{1}{X} = 1 - \left[\frac{v_g}{v_f}\right]^{.67}\left[1 - \frac{1}{\alpha}\quad (v_g/v_f)^{.1}\right]$$

which was developed by von Glahn to relate the exit quality and exit void fraction. Both the Lottes-Flinn[10] and the Martinelli-Nelson[11] two-phase friction multiplier correlations were used in the code. The other relationship utilized in the code will be described in the appropriate section.

1. Driving Pressure[9]

In a natural circulating system the force which causes flow is derived from the difference in density in the downcomer and riser loops of the systems. The feedwater in the downcomer is saturated or slightly subcooled and will have a density much greater than the density of the two-phase mixture in the riser loop. Because of this density difference, a driving pressure is established and is given by

$$DP = \left[\rho_{dc1} Z_1 + \rho_{dc2} Z_2 - \left(\bar{\rho}_o H_o + \bar{\rho}_b H_b + \rho_c H_c\right)\right]\frac{g}{g_c}$$

which is the difference of hydrostatic pressure in the downcomer and in the boiler.

2. Pressure Losses

In a natural circulating system equilibrium flow will exit when the driving pressure is equalled by the total pressure losses in the system. When the driving pressure is greater than the pressure loss in the system, the flow will increase causing greater frictional losses and the system will reach a new equilibrium flow. Conversely, if the pressure losses exceed the driving pressure, the flow decreased until a new equilibrium flow is established.

The pressure losses in the system consist of frictional losses in the channel, chimney and downcomer, acceleration pressure losses in the channel, and the pressure losses due to area contractions and expansions. The frictional pressure losses were calculated by the Darcy Formula[9]

$$\Delta P = \bar{R} f \frac{H}{D} \frac{\rho V^2}{Zgc}$$

where

f = friction factor

D_e = equivalent diameter, ft

$\bar{\rho}$ = average density, lb/ft^3

$\bar{V}$ = average velocity, ft/hr

g_c = conversion factor

$\bar{R}$ = two-phase friction multiplier

= 1 for single-phase flow.

The pressure change due to sudden expansions were calculated by[9]

$$\Delta P = \frac{1}{g_c}\left(\frac{1}{A_1A_2} - \frac{1}{A_2^2}\right)\frac{\dot{m}_t^2}{\rho}$$

for single-phase flow and by

$$\Delta P = \frac{1}{g_c}\left(\frac{1}{A_1A_2} - \frac{1}{A_2^2}\right)\dot{m}_t^2\left(\frac{(1-x)^2}{\rho_f(1-\alpha)} + \frac{x^2}{\rho_g\alpha}\right)$$

for two-phase flow. A_1 is the upstream cross sectional area and $\dot{m}_t$ is the total mass flow, lb/sec, in the channel. The pressure change due to sudden contraction was calculated by[9]

$$\Delta P = 0.0217\left(\frac{1}{A_1^2} - \frac{1}{A_2^2}\right)\frac{\dot{m}_t^2}{\rho}$$

for single-phase flow and

$$\Delta P = 0.0186\left(\frac{1}{A_1^2} - \frac{1}{A_2^2}\right)\frac{\dot{m}_t^2}{\rho_f}\ \frac{(1-x)^2}{1-\alpha}$$

for the two-phase flow.

The two-phase acceleration loss was calculated by[9]

$$\Delta P = \frac{G^2}{32.2}\left(\frac{(1-x_e)^2 v_f}{1-\alpha_e} + \frac{x_e^2 v_g}{\alpha_e} - v_i\right)$$

In the NWB the power density of the fuel is decreasing with time from 319 w/1 at startup to 50.6 w/1 when it is removed from the boiler. Therefore, the heat available for transfer to the two-phase mixture will decrease raising the average coolant density in the core. However, the density of the fluid in the downcomer will remain constant. This will decrease the driving pressure in the boiler and reduce the flow. In this analysis a constant exit quality model was assumed. By keeping the exit quality constant, the average density in the riser remains constant and the driving pressure is unchanged. However, the flow must be reduced in order to keep the exit quality constant as the power density decreases, and this results in decreased pressure losses.

The driving pressure and pressure losses of the system as functions of power density and time over the life cycle of the fuel in the NWB are shown in Figures 4-8, 9, 10 and 11. From these figures it is seen that the driving pressure exceeds the total pressure loss in the system at all times for both the Lottes-Flinn and the Martinelli-Nelson correlations. The figures were developed assuming a constant exit quality of 5%. To maintain this constant quality, the mass flow must decrease as the power density decreases. The flow control is maintained by introducing additional pressure loss in the downcomer. The control pressure loss required is the difference between the driving pressure and the total pressure loss.

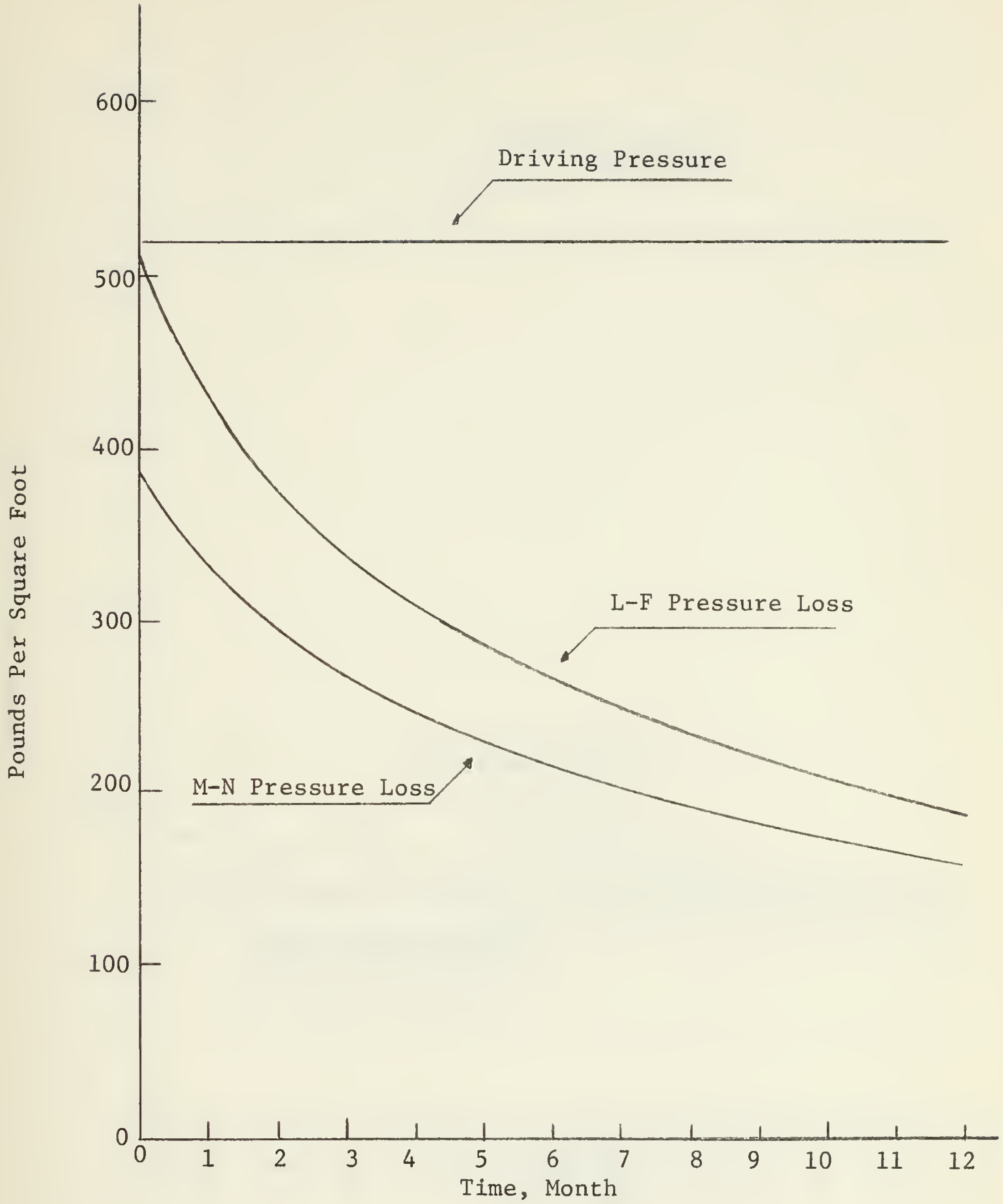


Figure 4-8

Driving Pressure and Pressure Losses
Fuel Element Age from 0 to 1 Year
Power Density from 320 to 147 W/L

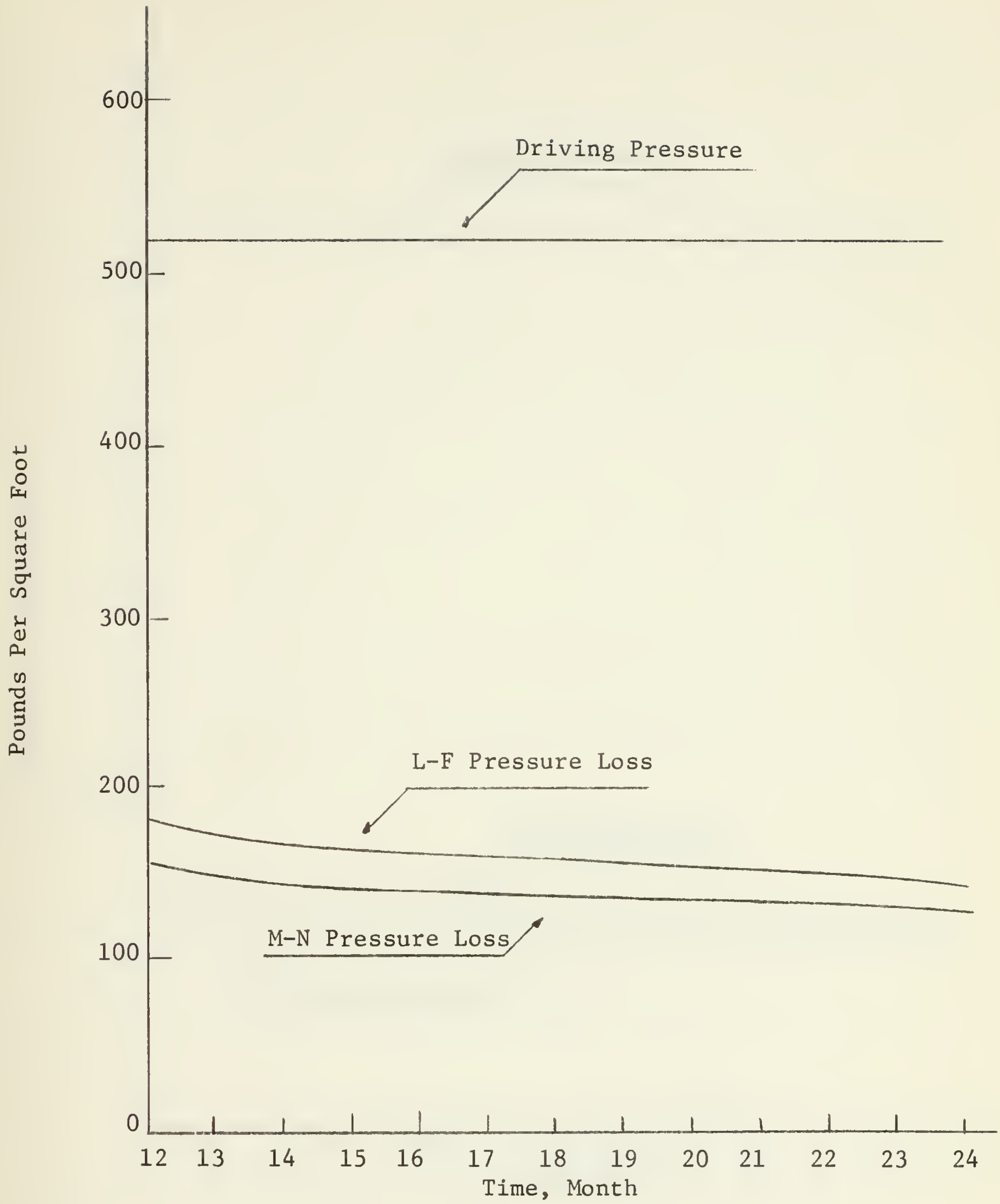


Figure 4-9

Driving Pressure and Pressure Losses
Fuel Element Age from 1 to 2 Years
Power Density from 147 to 102 W/L

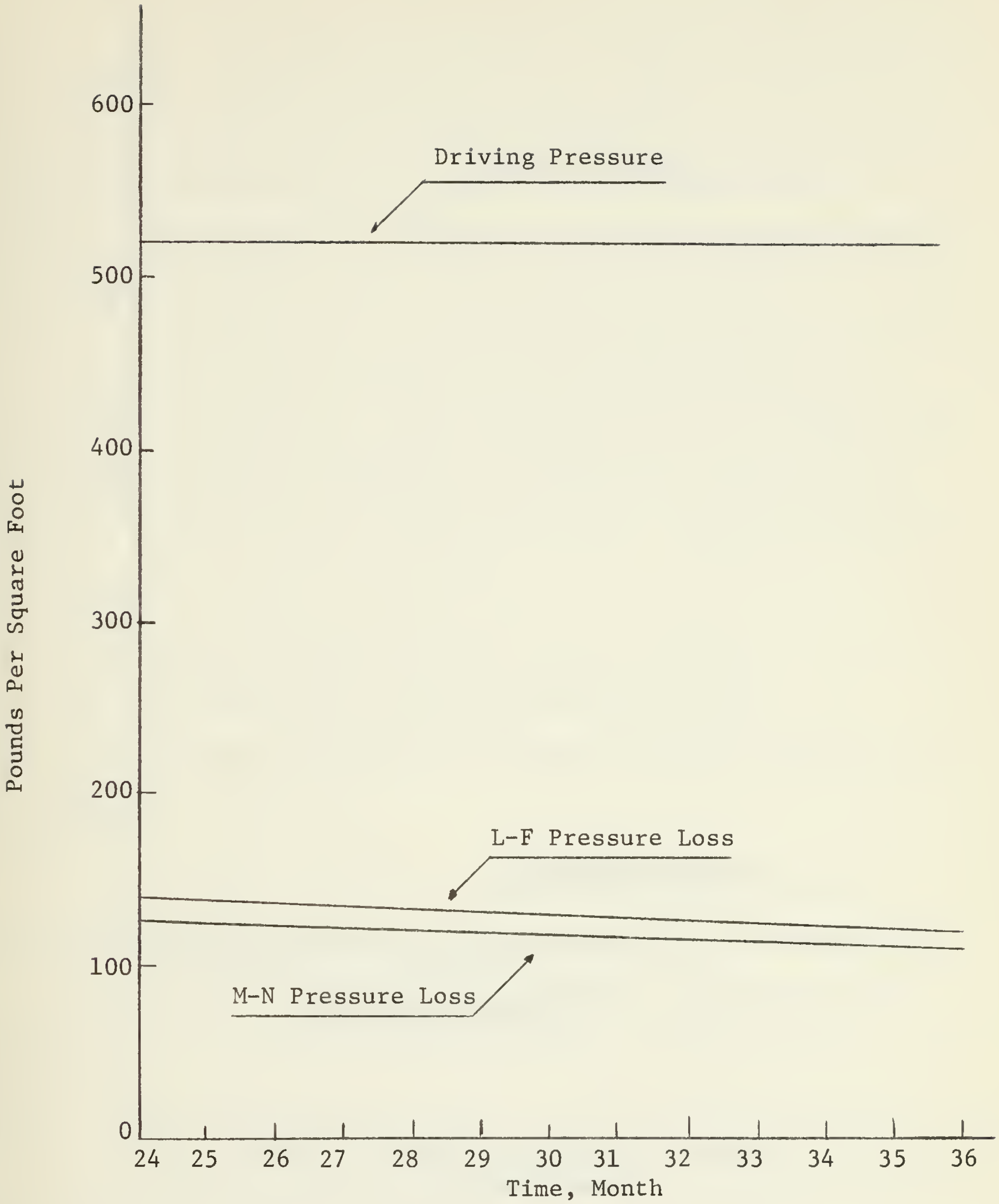


Figure 4-10

Driving Pressure and Pressure Losses
Fuel Element Age from 2 to 3 Years
Power Density from 102 to 64 W/L

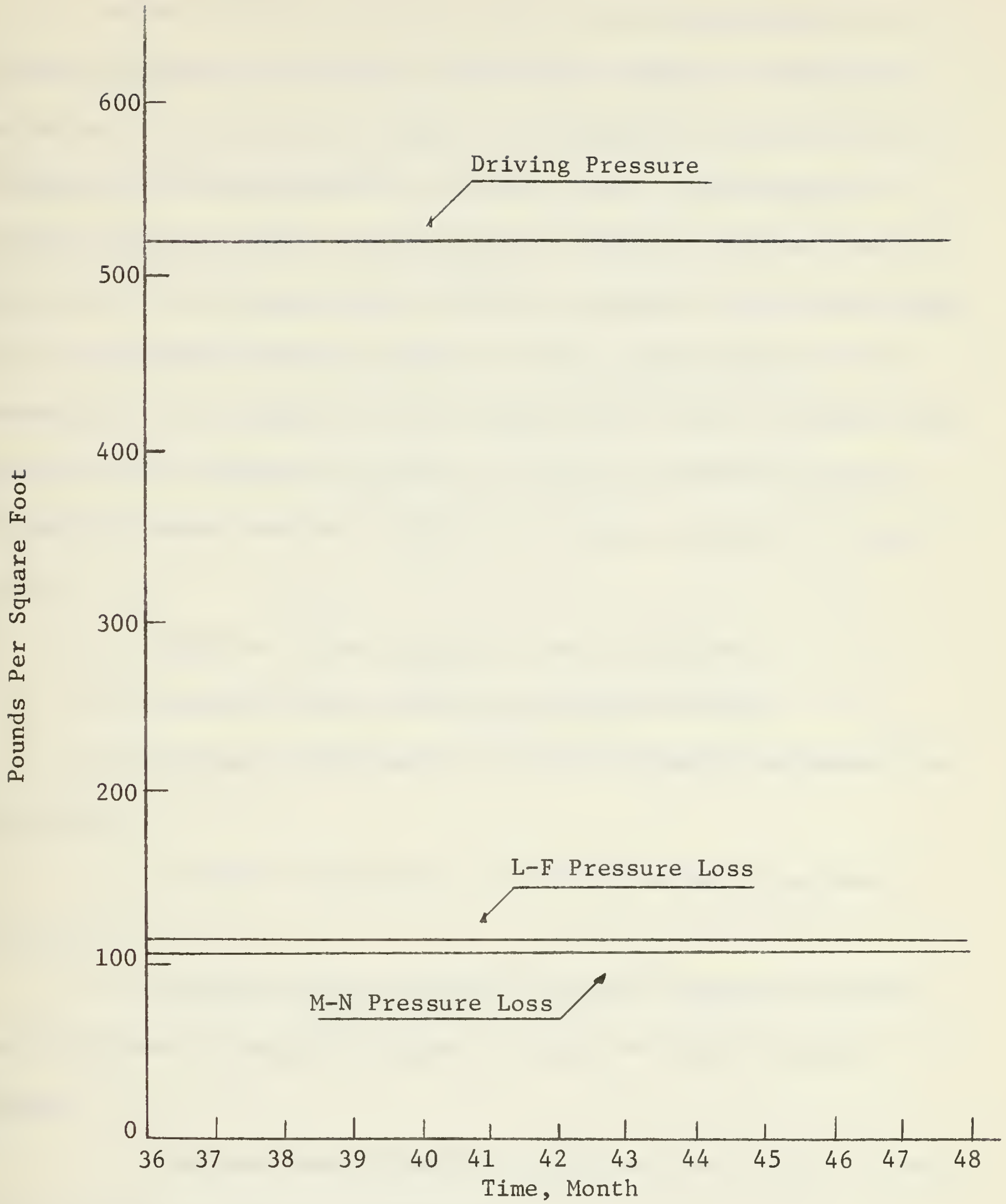


Figure 4-11

Driving Pressure and Pressure Losses
Fuel Element Age from 3 to 4 Years
Power Density from 64 to 50 W/L

3. Flow Stability[12]

In the natural circulating loop, the driving pressure is directly related to the void in the boiling channel. A temporary reduction of the inlet flow leads to an increase in the local void fraction and an acceleration in the exit flow. The increase in pressure drop may offset the increase in available driving pressure due to the void increase, and a further reduction of flow rate occurs with a consequent further increase in voids, which leads to flow instability. The subject of flow instability is far from being fully explored, but a summary of effects of various parameters of flow instability which have been observed in various studies are listed below:

a. High heat input aggravate flow instability.

b. Increased inlet subcooling reduce stability.

c. Increased resistance at the exit strongly decreases flow stability.

d. Increased resistance at the inlet strongly increases stability.

e. Increased system pressure increases stability because the difference in specific volumes of vapor and liquid becomes smaller.

f. Increased static head in a long vertical channel increases the stability of upward boiling flow.

To check the flow stability of this system, the pressure drop in the channel as a function of mass flow was determined. These calculations were performed for the maximum power density in each boiler

section and are shown in Figures 4-12a, b, c and d. The pressure losses were calculated using both the Lottes-Flinn and the Martinelli-Nelson correlations as shown in the figures. Also the normal operating mass flow for each power density is indicated. It is seen that stable conditions exist in each case. A temporary reduction in flow causes a reduction in pressure loss which will tend to increase the driving pressure and return the flow to normal. Associated with an increase in flow is an increase in pressure loss which will decrease the driving pressure decreasing the flow to normal. Additionally, the sensitivity of the Lottes-Flinn correlation to increased void fraction is shown in Figure 4-12a. At the relatively high heat rate of 320 w/l the quality and void fraction increase with decreased flow, causing a rapid increase in two-phase friction losses, which is indicated by the sharp increase in pressure loss in the channel. The heat rates in the remaining figures were too low to show this increase for the flow rates calculated.

4. Cross Flow

The flow through the NWB core is in parallel channel around the fuel elements. This open matrix core is connected at the channel inlet and exit by open plena. This will cause the pressure at the inlet for all channels to be a constant, P_i, and the pressure at the exit of all channels to be another constant, P_e. The flow in each channel is dependent upon the channel area and the heat rate in the channel. All channels are assumed to be equal and will not effect the flow. However, a difference in heat rate in adjacent channels will result in a cross flow between channel.

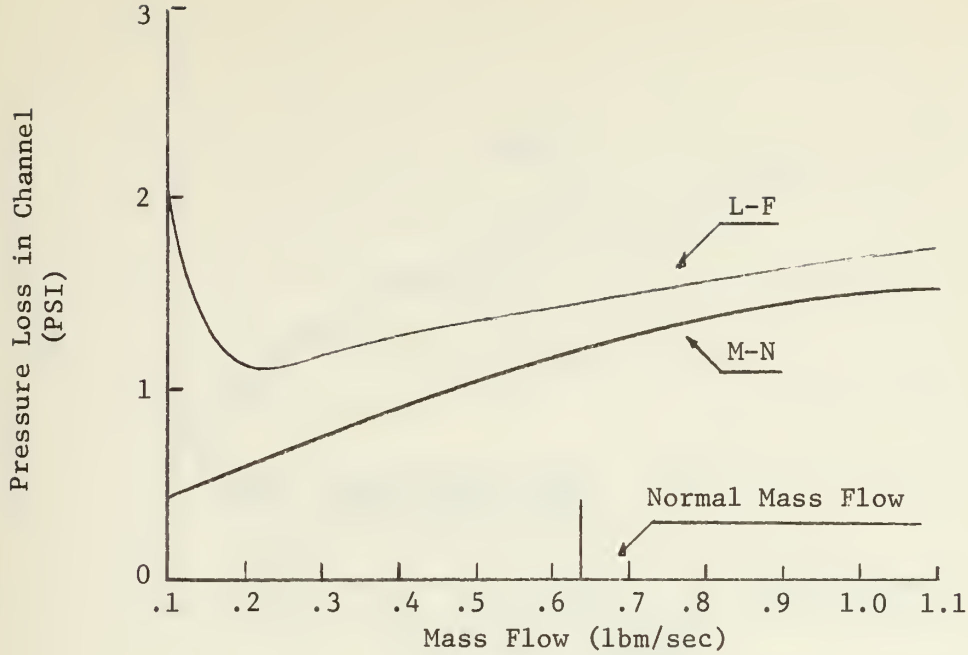


Figure 4-12a

Channel Pressure Loss at 320 W/L

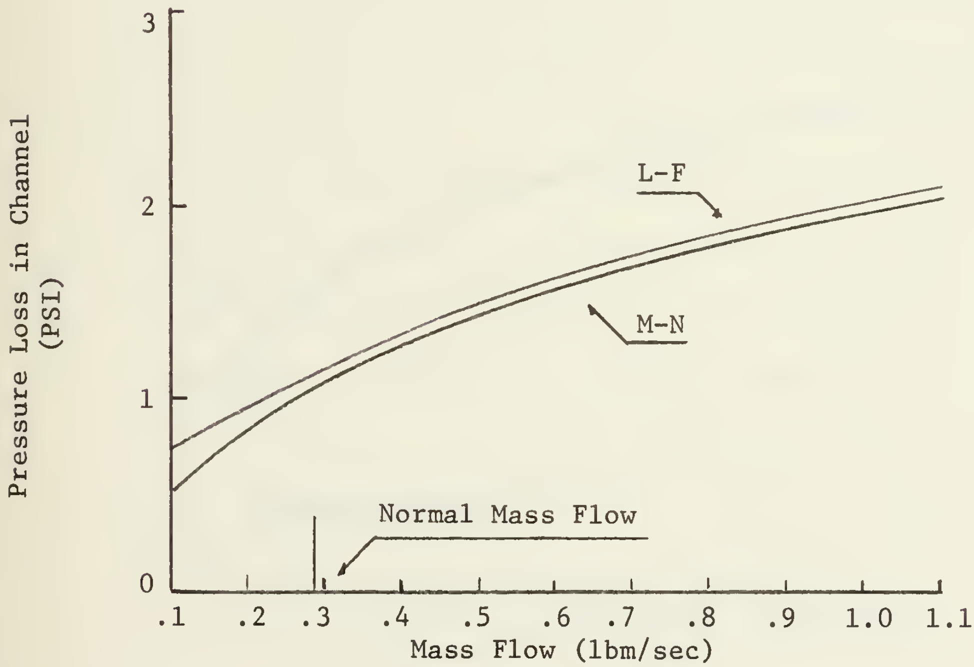


Figure 4-12b

Channel Pressure Loss at 140 W/L

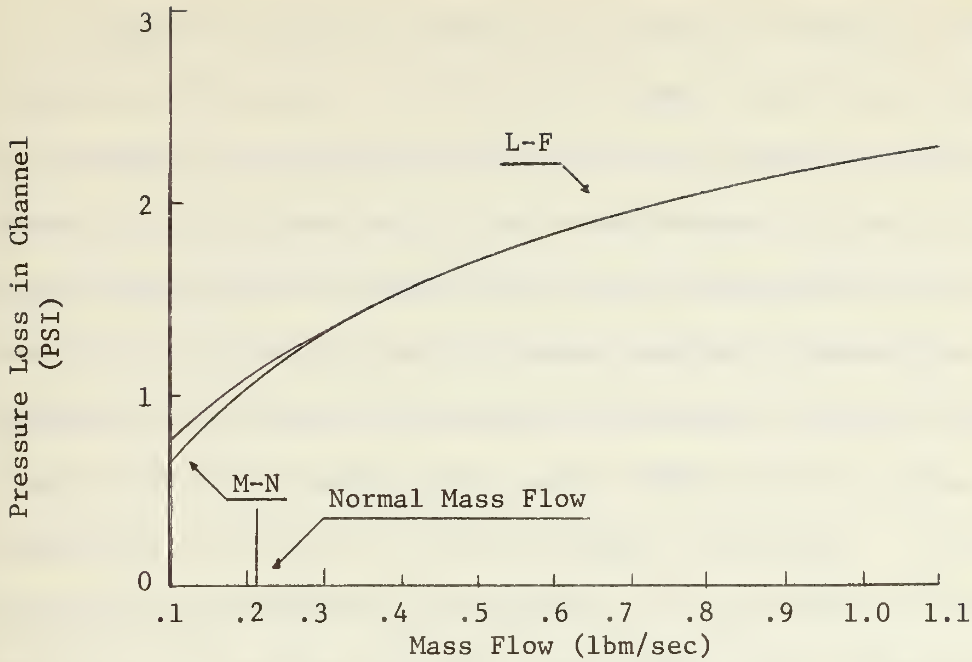


Figure 4-12c

Channel Pressure Loss at 99 W/L

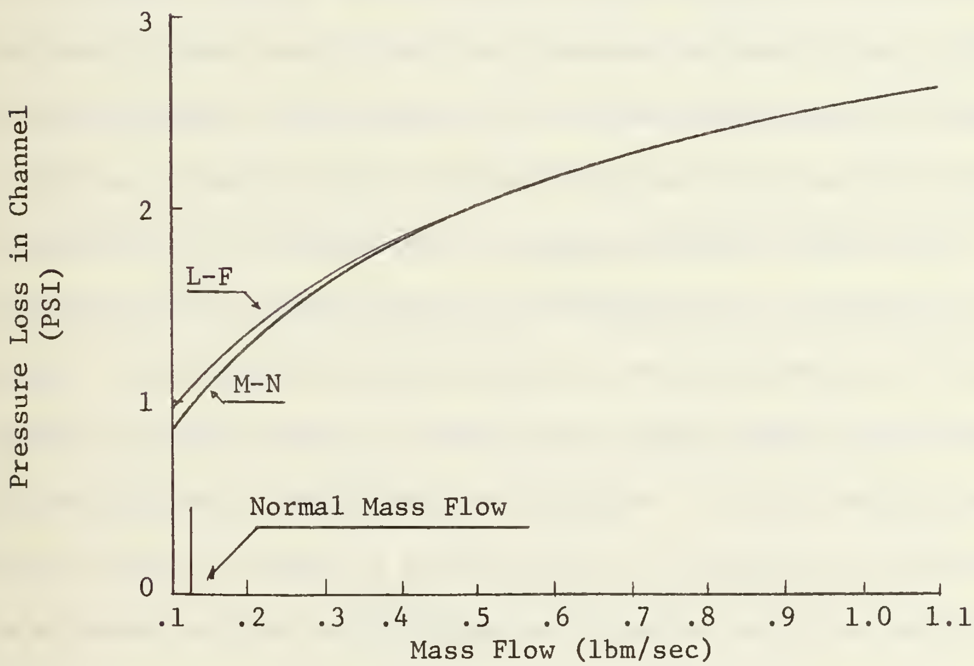


Figure 4-12d

Channel Pressure Loss at 63 W/L

When two adjacent channels are heated at different rates, a cross flow will result. In the hotter channel, more heat is produced for transfer to the water and will cause the coolant in the hot channel to boil before that of the cooler channel. Since the two-phase pressure loss is greater than that of single-phase, the pressure loss will be greater in the hot channel causing a greater flow resistance. To equalize the pressure difference, a part of the hot channel flow will migrate to the cooler channel and will cause the temperature of the fuel in the hot channel to increase. If the cross flow from the hot channel is sufficient, it is possible to raise the temperature of the fuel element to the melting point. Therefore, to insure fuel integrity, a cross flow analysis should be performed.

Due to the time required to accumulate enough fuel elements for a boiler section, and the decay characteristic of the fuel, a variance of heat rate in the fuel elements is possible and cross flow is probable. Therefore, a cross flow analysis was performed utilizing a computer code developed by Atomic International, and obtained from the Code Center at Argonne National Laboratory[13]. This code uses a multi-channel, two-dimensional, two-phase flow model. It assumes that the pressure in every channel is the same across the axial station. The pressure drop between axial stations in each channel is calculated and compared. If the pressure drops are not equal to within a given value, the mass flow in each channel is adjusted and the calculations are repeated until the relative error is acceptable. For this analysis, the worst possible case of adjacent channels with 340 and 260 w/l heat rates was used.

The results are shown in Figure 4-13. From this figure, it is seen that no cross flow exists until boiling begins in the hot channel. At that point, the flow in the cooler channel increases until boiling occurs in the cooler channel. When both channels are boiling, the rate of cross flow decreases and returns to equal flow in both channels. The maximum flow reduction in the hot channel is less than 10%. As will be shown in Chapter VI, this flow reduction will not result in excessive temperatures in the fuel.

C. Fuel Storage Pool

Plan and cross section elevation views of the fuel storage pool are shown in Figures 4-14 and 4-15. The storage pool is divided into two sections. Section A is large enough to store all the fuel in the NWB. Normally it will not contain fuel, but will provide a heat sink for the heat removal system. If repairs to the boiler are required, all fuel elements can be stored in this area. Section B will be used to store the fuel elements when they are received from the reprocesser or when they are waiting shipment after removal from the boiler. It is large enough to hold the fuel elements from two boiler sections and it also contains an isolation cell where the fuel elements are removed from their shipping containers and checked for leaks.

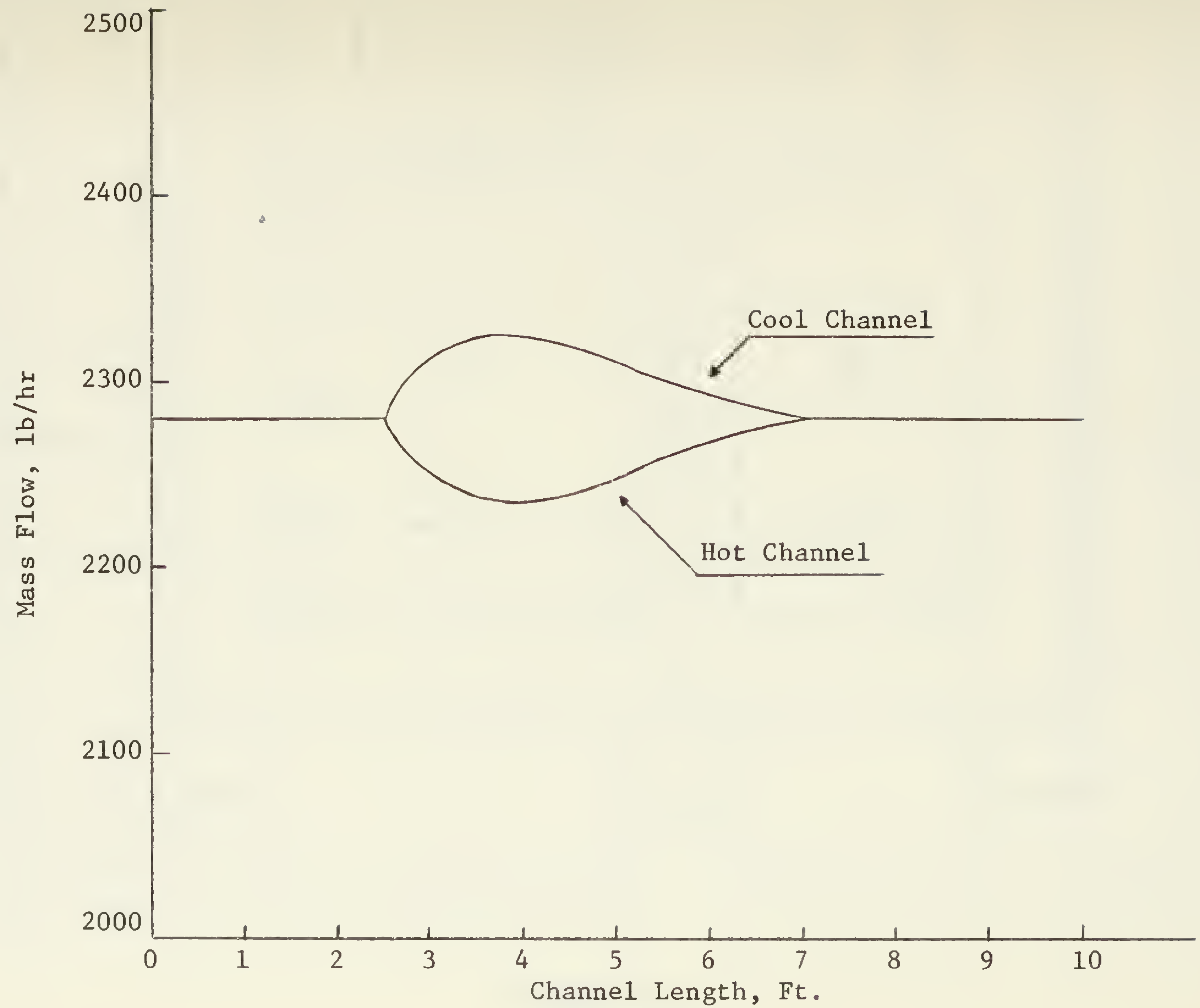


Figure 4-13

Cross Flow Between Hot and Cool Channels

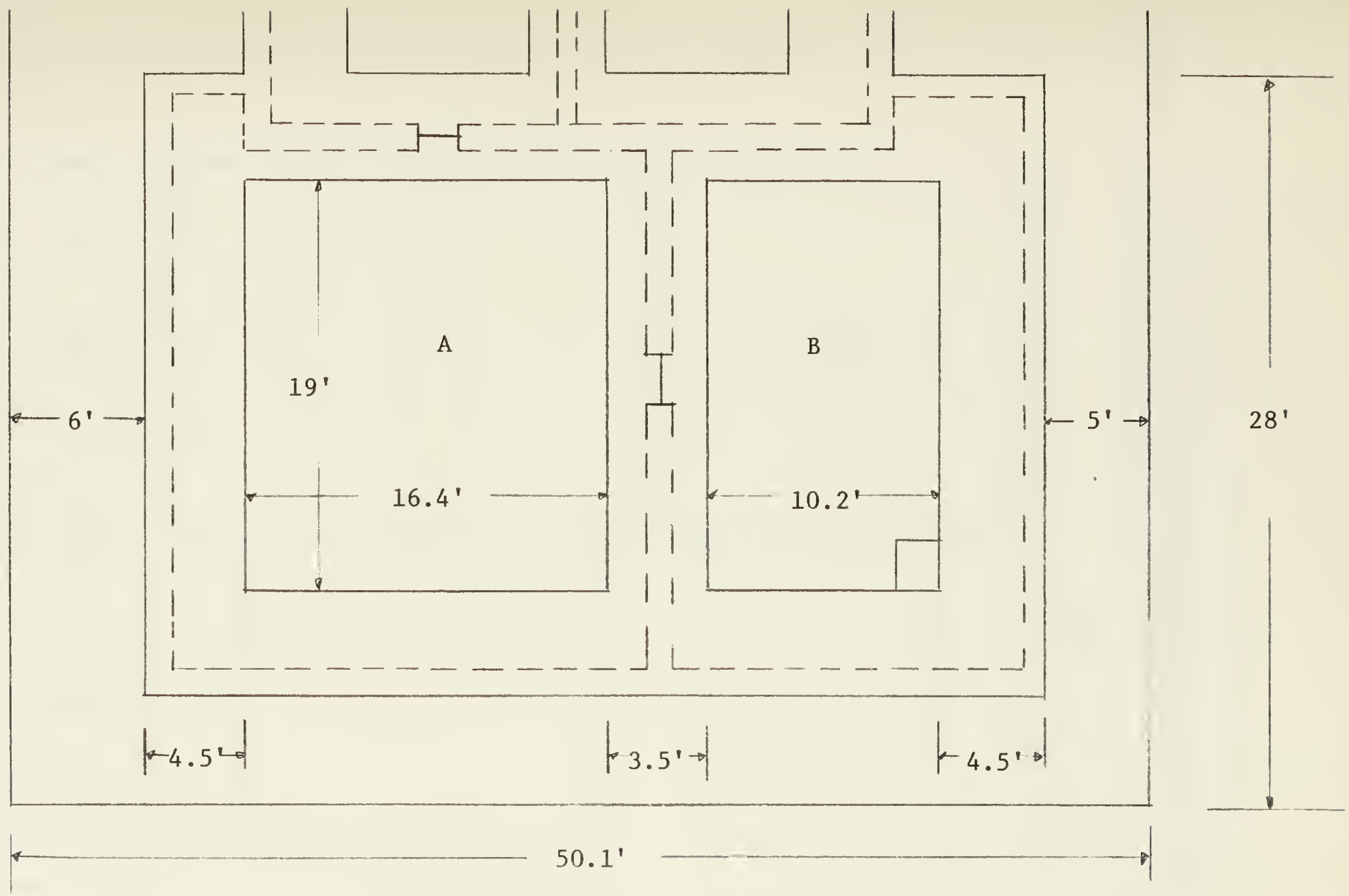


Figure 4-14

Plan View of Fuel Storage Pool

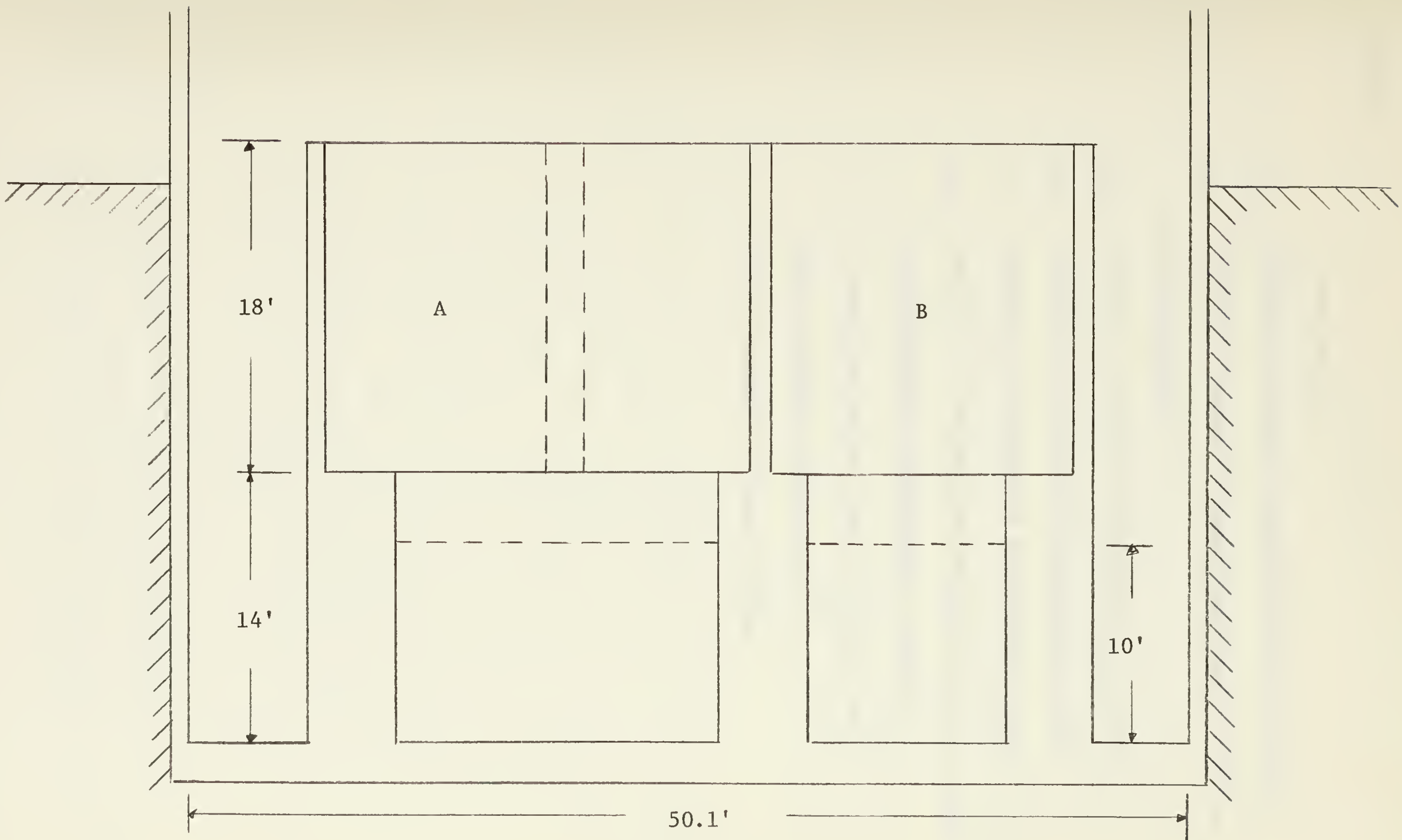


Figure 4-15

Elevation View of Fuel Storage Pool

Glossary

X	=	Steam quality
v_g	=	Specific volume of saturated vapor, ft^3/lbm
v_f	=	Specific volume of saturated liquid, ft^3/lbm
α	=	Void fraction
ρ_{dc1}	=	Density of fluid in downcomer Z_1, lb/ft^3
ρ_{dc2}	=	Density of fluid in downcomer Z_2, lb/ft^3
ρ_o	=	Density of coolant in non-boiling height H_o, lb/ft^3
ρ_b	=	Density of coolant in boiling height H_b, lb/ft^3
g	=	Gravitational acceleration, ft/sec^2
g_c	=	Conversion factor, $lbm - ft/lb_f - sec^2$
G	=	Mass velocity, $lb/sec - ft^2$
v_i	=	Inlet specific volume, ft^3/lbm

V. SHIELDING

The nuclear waste used as fuel elements in the NWB will be received in a shipping container which will provide adequate radiation protection during transit. The shipping container will be submerged in the storage pool before the fuel element is removed. The fuel will be removed under water and normally will remain under water during storage, transfer and use. When the fuel is readied for shipment to final storage, it will be returned to a shipping cask before it is removed from the storage pool. In the design of the NWB, it is important to insure that adequate biological protection is provided to reduce the radiation to acceptable levels when the fuel is not in the transfer cask.

The solidified nuclear waste material is contained in stainless steel cylinders. The attenuating effects of the stainless steel cylinder and the stainless steel pressure vessel liner have been neglected in the calculations. Additionally, the fuel will be approximately 90 days old at the time it is delivered to the boiler plant. The activity level at 90 days following reprocessing was used in the calculations. These simplifications are conservative and tend to over estimate the actual dose rates. Also, they are applicable immediately after refueling and will decrease as the fuel decays.

The shielding design was based on the limiting dose rates published in 10CFR20. This limiting dose is 1-1/4 rems per calendar quarter to the whole body, head and trunk, active blood forming organs,

lenses of the eyes, or gonads. This is equivalent to a dose rate of 100 millirems in a seven day week.

A. Calculation Models[14]

To determine the shielding requirements for the Nuclear Waste Boiler (NWB), a computer code was developed to calculate the dose rate as a function of shield thickness. The code was based on the semi-space volume source and the cylindrical volume source models.

1. Semi-Space Uniform Volume Source

The semi-space uniform volume source model is shown in Figure 5-1.

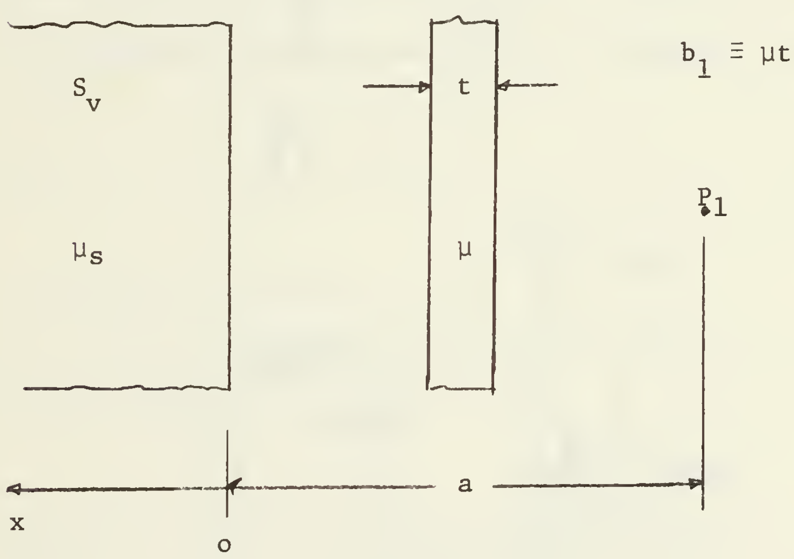


Figure 5-1

Semi-Space Uniform Volume Source Model

The flux at point P_1, a distance a from the semi-space uniform volume source is given by

$$\phi = \frac{BS_v}{2\mu_s} E_2(b_1)$$

where

B = buildup factor

S_v = constant volume source strength

μ_s = source linear attenuation coefficient

$E_2(b_1)$ = exponential integral function

2. Cylindrical Volume Source

The cylindrical volume source model for a self-absorbing medium is shown in Figure 5-2.

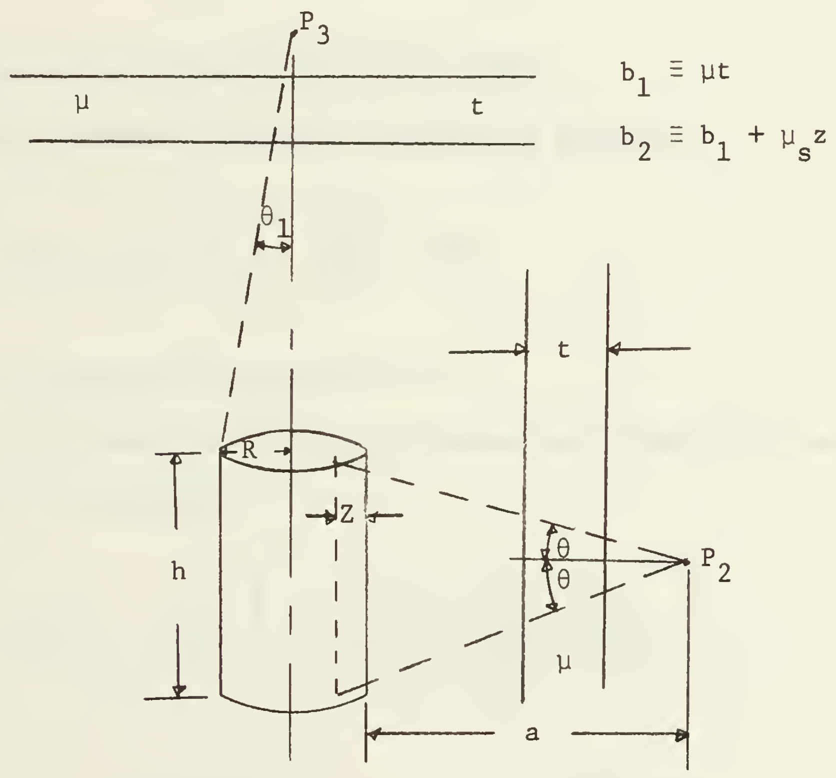


Figure 5-2

Cylindrical Uniform Volume Source Model

The flux at point P_2 , a distance a from the cylindrical volume source is given by

$$\phi = \frac{BS_v R^2}{2(a+z)} \quad F(\phi, b_2)$$

where

z = self-attenuation distance

$F(\phi, b_2)$ = Secant integral function

The flux at point P_3 is given by

$$\phi = \frac{BS_v}{2\mu_s} \left[E_2(b_1) - \cos\theta_1 \, E_2\,(b_1 \sec\theta_1) \right]$$

3. Exponential Integral Function, $E_n(b)$

The exponential integral function is defined by

$$E_n(b) \equiv b^{n-1} \int_b^{\infty} \frac{e^{-x}}{x^n} \, dx$$

where n is an integer ≥ 0 and $b \geq 0$.

For values of b from 1 to 110 the following approximation is accurate to one part in a million

$$E_1(b) = \frac{e^{-b}}{b} \left[\frac{a_o + a_1 b + a_2 b^2 + b^3}{c_o + c_1 b + c_2 b^2 + b^3} \right]$$

where

a_o = 0.2372905 $\quad$ c_o = 2.4766331

a_1 = 4.5307924 $\quad$ c_1 = 8.6660126

a_2 = 5.1266902 $\quad$ c_2 = 6.1265272

As used in the calculation models the exponential integral function is $E_2(b)$. Using the reduction formula,

$$E_n(b) = \frac{1}{n-1}\left[e^{-b} - bE_{n-1}(b) \right] \quad \text{for } n > 1, \text{ we get}$$

$$E_2(b) = e^{-b} - bE_1(b)$$ and the above approximation may be used.

4. Secant Integral Function, $F(\theta,b)$

The secant integral function is defined by

$$F(\theta,b) \equiv \int_0^\theta e^{-b \sec \theta}\, d\theta \quad \text{for } b \geq 0 \text{ and } \frac{\pi}{2} \geq \theta \geq 0.$$

In the shielding calculations the approximation

$$F(\theta,b) \simeq \theta\, e^{-b}$$

for $\theta < 5°$ in radians and

$$F(\theta+\delta,b) \simeq F(\theta,b) + \delta e^{-b \sec \theta}$$

for δ in radians was used.

5. Dose Rate

The relationship between the absorbed dose rate and the energy flux density is given by

$$DR = 1.6 \times 10^{-8}\, \phi\, E\, \overline{\mu}_a \quad \text{rad/sec}$$

where

ϕ = flux, photons/cm^2 - sec

E = energy, MeV

$\overline{\mu}_a$ = mass absorption coefficient of γ rays in air, cm^2/gm

In terms of mr/hr, the dose rate is given by

$$DR = 0.0576\ \phi\ E\ \overline{\mu}_a \text{ mr/hr}$$

In the shielding calculations, the photon energies were averaged into four energy groups. To determine the total dose rate at a point it is necessary to sum the dose rates for each energy group.

B. Volumetric Source Strength, S_v

The volumetric source strength, S_v photons/cm^3 - sec, is dependent on the radionuclides present in the waste and their concentrations. Table 5-1 shows the isotopic data of the radionuclides normally present in the typical high level waste[15]. In reference 15, Renzetti performed an extensive analysis of the radionuclides present in the waste. In this analysis, it was determined that the estimated bremsstrahlung photon energy was 0.05 MeV, and was less than the minimum gamma energies listed in Table 5-1, and could be neglected in the shielding calculations. Also, the significant gamma contributors range from .272 to 2.63 MeV and can be divided into four energy groups for ease of calculations. This division of radionuclides is summarized in Table 5-2.

Table 5-1

Isotopic Data of Nuclides Normally Present In High-Level Waste

Nuclide	Major Radiations Energies, MeV Intensities, %		Half-Life	Decay Constant $Year^{-1}$
Sr-90	B	0.546 (max)	27.7 yr	0.025
Y-90	γ	1.75 (0.2%)	64 hr	0.025
	B^-	2.27 (max)		
Zr-95	γ	0.724 (49%)	65 d	0.18
		0.756 (49%)		
		0.235 (2%)		
Nb-95	γ	0.765 (100%)	35 d	7.22
Tc-99	B	0.292 (max)	2.12×10^5 yr	3.3×10^{-6}
Ru-106	B	0.039 (max)	368 d	0.693
Rh-106	γ	2.37 (0.1%)	30 s	0.693
		2.63 (0.1%)		
		2.44 (0.1%)		
		2.3 (0.1%)		
		1.56 (1%)		
		1.13 (11%)		
		0.512 (8%)		
Sb-125	γ	0.462 (43%)	2.7 yr	0.26
		0.598 (28%)		
		0.604 (7%)		
Te-125m	γ	0.109 (100%)	58 d	4.3
Te-127m	B^-	0.73 (max)	109 d	2.3
Te-127	B^-	0.70 (max)	9.4 h	
Cs-134	γ	1.37 (27%)	2 yr	0.35
		1.04 (2.5%)		
		0.796 (71%)		
Cs-137	B^-	0.514 (max)	30 yr	0.023
Ba-137m	γ	0.662 (100%)	2.6 m	0.023
Ce-144	B^-	0.031 (max)	285 d	0.89
Pr-144	γ	0.695 (1.5%)	17.3 m	0.89

Table 5-1 (cont.)

Nuclide	Major Radiations Energies, MeV Intensities, %		Half-Life	Decay Constant Year^{-1}
		2.19 (1%)		
	B^-	2.99 (max)		
Pm-147	B^-	0.224 (max)	2.6 yr	0.26
Sm-151	B^-	0.076 (max)	87 yr	$7.85\ 10^{-3}$
Eu-154	γ	1.6 (29%)	16 yr	0.04
		1.5 (2%)		
		1.28 (37%)		
		1.01 (17%)		
		1.00 (2%)		
		0.817 (3%)		
		0.371 (10%)		
	B^-	1.85 (max)		
Eu-155	B^-	0.25 (max)	1.8 yr	0.385
Np-239	γ	0.334 (32%)	2.4 d	
		0.286 (40%)		
		0.272 (7%)		
	B^-	0.273 (max)		
Pu-238,9, 40,41	α			
Am-241	α			
Cm-242	α			

Table 5-2

Radioactivity of Major Gamma Emitting Nuclides
In Typical High-Level Waste by Energy Group

		Radioactivity, Ci/MT			
		Energy Group, MeV			
Nuclide	Total Radio-Activity	0.395	0.785	1.46	2.39
Y-90	74,800			15	
Zr-95	14,300		14,000		
Nb-95	72,000		72,000		
Rh-106	246,000	20,900		20,500	1,000
Sb-125	6,650	2,900	2,500		
Tc-127	600	Negligible			
Cs-134	167,000		118,600	48,400	
Ba-137m	97,300		97,300		
Pr-144	395,000		5,900		3,950
Eu-154	6,550	700	1,400	4,500	
Np-239	18	Negligible			
	Total	24,500	311,700	83,400	5,000

From reference 2, one metric ton of reactor fuel produces $3.3\ ft^3$ of waste. The volume of waste in each NWB fuel element is $3.49\ ft^3$. Therefore, each NWB fuel element contains the waste from 1.0576 MT of reactor fuel or each section of the boiler contains the wastes produced from 178.73 MT of reactor fuel. The volumetric uniform source strength for each energy group is given by

$$S_V(E) = \frac{FA(E)}{V}$$

where

$A(E)$ = activity by energy group, Ci/MT

V = volume of source, $2.545 \times 10^7\ cm^3$

F = factor = $178.73\ MT \times 3.7 \times 10^{10}$ dis/sec/Ci.

Simplified

$$S_V(E) = 2.5984 \times 10^5\ A(E)\ photon/cm^3 - sec.$$

The energy grouped volumetric source strengths are given in Table 5-3.

Table 5-3

Volumetric Source Strength of High-Level Solid Waste Per Boiler Section

Energy (MeV)	Sv (photons/cm^3 - sec)
0.395	6.366×10^9
0.785	8.099×10^9
1.46	2.167×10^{10}
2.34	1.299×10^9

C. Shielding Parameters

1. Linear Attenuation Coefficients, μ_m

The linear attenuation coefficients for concrete at a density of 2.3 gm/cm^3 and water at a density of 1 gm/cm^3 were obtained from reference 14, and are listed by energy groups in Table 5-4.

Table 5-4

Linear Attenuation Coefficients of Concrete and Water

	μ, cm^{-1}	
Energy, MeV	Concrete	Water
0.395	0.220	0.106
0.785	0.163	0.079
1.46	0.120	0.057
2.39	0.095	0.045

2. Buildup Factor, B

The buildup factor can be determined by

$$B = A_1 e^{-\alpha_1 \mu x} + A_2 e^{-\alpha_2 \mu x}$$

where the constant depends upon the shielding material and photon energy. The values of these constants were obtained from reference 16 and are listed in Table 5-5.

Table 5-5

Buildup Factor Constants for Concrete and Water

Energy, MeV	A_1	A_2	α_1	α_2
		Concrete		
0.395	12.5	-11.5	-0.115	0.005
0.785	10.5	- 9.5	-0.091	0.024
1.46	7.5	- 6.5	-0.075	0.048
2.39	5.5	- 4.5	-0.065	0.065
		Water		
0.395	24.0	-23.0	-0.138	0.0
0.785	11.7	-10.7	-0.108	0.0
1.46	7.8	- 6.8	-0.087	0.075
2.39	5.8	- 4.8	-0.068	0.098

3. Linear Attenuation Coefficient in Source

The linear attenuation coefficient of the source material may be expressed as

$$\mu_s = N\sigma_t$$

where the total miscroscopic cross section, σ_t, is the sum of the Compton scattering, σ_c, photoelectric interaction, σ_{ph}, and the pair production, σ_{pp}, cross section.

The source material is composed of a homogeneous mixture of about 30 elements. To calculate the linear attenuation coefficient, it is first necessary to determine the atom density of each element of the mixture. This analysis was performed in reference 15 and the results are shown in Tables 5-6 and 5-7.

Table 5-6

Atom Densities of Radionuclides in Solid Waste

Nuclide	Atom Density, atoms/cm^3
Sr-90	7.3×10^{20}
Y-90	7.3×10^{20}
Ru-106	9.4×10^{19}
Rh-106	9.4×10^{19}
Cs-134	1.2×10^{20}
Cs-137	1.1×10^{21}
Ba-137	1.0×10^{21}
Ce-144	1.1×10^{20}
Pr-144	1.1×10^{20}
Pm-147	6.8×10^{19}
Sm-151	2.6×10^{19}
Eu-154	3.9×10^{19}

Table 5-7

Macroscopic Cross Section of the Source Material

Energy, MeV	μ_s, cm^{-1}
0.395	0.136
0.785	0.224
1.46	0.228
2.39	0.339

4. Self-Attenuation Distance, z

The self-attenuation distances were obtained from reference 16 and are summarized in Table 5-8 below.

Table 5-8

Effective Self-Attenuation Distance

Energy	Z
0.395	4.4
0.785	5.6
1.46	4.9
2.39	4.9

D. Results

1. Nuclear Waste Boiler

In the NWB the fuel elements are placed in a closely spaced lattice which can be approximated by a uniform volume source. The semi-space uniform volume source model was used to calculate the dose

rates under normal operation and during refueling. During refueling, the post-tensioned concrete cover will be removed from the section being refueled and the shielding will be provided by the water cover only. The situation is identical to the normal storage pool condition and will be discussed later.

The results of the shielding calculation are shown in Table 5-9 and Table 5-10.

Table 5-9

Concrete Thickness and Corresponding Dose Rate at Exterior Surface of Boiler Wall, Volume Source

Thickness, Ft.	Dose Rate, Mr/hr
3.5	488
3.75	213
4.0	94
4.25	42
4.5	18.8
4.75	8.5
5.0	3.9
5.25	1.8
5.5	0.8

Table 5-10

Vertical Depth of Water and Corresponding Dose Rate at Surface of the Water, Volume Source, 12" Concrete Included

Depth, Ft.	Dose Rate, Mr/hr
6	195
7	41.7
8	9.2
9	2.1
10	0.5

Structurally, 3.5 feet of reinforced concrete will more than adequately support the pressure vessel design load[15]. Therefore, the biological shielding requirement will determine the boiler wall thickness. It is expected that personnel will be in the area of the boiler to perform minor maintenance, testing, and inspection of the vessel walls. The only components which require maintenance are the motorized gate valves. These valves are normally fully open or fully closed, and will require little maintenace, but must be tested periodically. It is estimated that the repairs around the boiler can be completed in less than four hours. Based on the four hours exposure time, a wall thickness of 4.5 feet was selected for the exterior pressure vessel wall. The dose rate at the exterior surface of the boiler at startup is 18.8 mr/hr. This allows a man to work at the boiler wall over five hours a week without exceeding the maximum allowable dose of 100 mr/week.

There are no mechanical components within the pressure vessel which require repairs or testing. Because of this and the fact that the interior of the pressure vessel is not readily accessible, an interior wall thickness of 3.5 feet between boiler sections was selected. If extended repairs to the fuel element support structures is required, portable shields can be used to reduce the radiation intensity and increase the allowable exposure time.

Based on Table 5-10, ten feet of water covering the boiler will reduce the dose rate to 0.5 mr/hr at the water surface. This level was selected as the minimum cover because it is anticipated that the water level must be sufficiently higher to provide adequate protection during fuel element transfer.

2. Storage Pool

The uniform volume source model was also used to calculate the dose rate for the storage pool. The condition for the pool wall is the same as the boiler wall, and therefore, the pool wall thickness of 4.5 feet was selected.

Above the storage pool, as above the boiler during refueling, the radiation attenuation is provided by a water cover. The uniform volume source model was used and the results are shown in Table 5-11.

Table 5-11

Vertical Depth of Water and Corresponding Dose Rate
At the Surface of the Water, Volume Source

Depth, Ft.	Dose Rate, mr/hr
8	230
9	49
10	10.8
11	2.4
12	0.57
13	0.13
14	0.03
15	0.008
16	0.002
17	0.001
18	0.0001

Before selecting the water depth of the storage pool, it is necessary to determine the required water cover during fuel element transfer between the storage pool and the boiler. For these calculations the cylindrical volume source model was used. For the horizontal water depth calculations, the attenuation of the one foot concrete retaining wall was included. The results are shown in Table 5-12 and Table 5-13.

Table 5-12

Vertical Depth of Water and Corresponding Dose Rate
At the Surface of the Water, Cylindrical Volume Source

Depth, Ft.	Dose Rate, mr/hr
5.5	263
5.75	164
6.0	103
6.25	65
6.5	41
6.75	26
7.0	16.8
7.25	10.8
7.5	6.9
7.75	4.5
8.0	2.9

Table 5-13

Horizontal Depth of Water and Corresponding Dose Rate
At Exterior Surface of Pool Water, Cylindrical Volume Source,
Include 12" Concrete Wall

Depth, Ft.	Dose Rate, mr/hr
4.5	1302
5.0	516
5.5	208
6.0	85
6.5	35
7.0	15

Based on the above results, seven feet of water cover for the transfer fuel element was selected as a minimum. This will result in a dose rate of 16.8 mr/hr at the water surface. Also, a minimum of six feet of horizional water thickness was selected and will result in a dose rate of 85 mr/hr at the refueling pool wall. The pool wall dose rate will increase above this high value when the fuel elements near the outer edge of the boiler are replaced because this minimum cannot be maintained. Normally, the refueling process will be carried out remotely with all personnel in the shielded control room during refueling to keep exposure to a minimum.

The normal water cover in the storage pool is determined by adding the ten foot length of the fuel element and a one foot clearance at the base of the element to the minimum cover during transfer. This results in an 18 foot cover and a dose rate of 0.0001 mr/hr at the water surface. During normal operation there will be negligible radiation on the second level of the NWB plant. Except during the refueling process, all radiation is confined to the pipe trench area surrounding the NWB and storage pool on the first level. During refueling the radiation on the second level will increase, but this area will be closed during actual fuel transfer.

VI. SAFETY

The high-level nuclear waste used as fuel in the NWB is primarily contained as a solid within a stainless steel cylinder. A secondary containment consists of a massive reinforced concrete structure. This secondary containment has penetrations which allow the coolant to circulate within the boiler plant during normal operation. However, there is an additional containment provided by the absorption unit, hot water converter or heat exchanger, where the steam is condensed and returned to the boiler. The only circulation outside of the boiler plant is the chilled water and the condenser water from the absorption units and the hot water from the converters or heat exchangers. See Figure 1-3. With these numerous barriers, it is highly unlikely that any radioactive contamination can spread outside of the boiler plant. In performing the system safety analysis, emphasis was placed on the integrity of the various containment barriers and the mechanism which will cause them to fail.

A. Fuel Temperatures

Failure of the fuel element itself can be due to rupture from a fall, corrosion of the stainless steel cylinder, or melting of the fuel cladding. Failure of the fuel element due to a fall was not considered as a problem. Shipping casks for the wastes are designed for a fall up to 30 feet. The maximum fall possible in the NWB plant is approximately five feet. Also, when the fuel element is out of the shipping cask, it will be under water, and any

accidental shock due to a fall in this case would be cushioned by the water. In either case, the fuel element will be isolated and tested. Any failed cannisters will be returned to the reprocessor. The remaining failure mechanisms are related to fuel temperature, and will be discussed in this section. According to reference 4, the important temperatures concerning the fuel elements are 1652°F for the fuel material, which is the melting point of the fuel, and 800°F for the cladding, which is the temperature above which there will be excessive corrosion of the cladding.

A detailed analysis of the fuel temperatures, under both normal operating and loss of flow conditions, was performed. The temperature relationship used in the calculations are outlined below[17].

1. The coolant temperature

$$t_f(\ell) = T_i + \frac{a_s \int_o^{\ell} q'''(\ell)}{\dot{m}C_p}$$

where

a_s = cross sectional area of the fuel, ft^2

q''' = volumetric heat source, $BTU/hr\text{-}ft^3$

$\dot{m}$ = mass flow rate, lbm/hr

C_p = specific heat, BTU/lbm-°F

Ti = inlet coolant temperature, °F

2. The surface temperature of the fuel element

$$t_s(\ell) = t_f(\ell) + \frac{a_s\, q'''(\ell)}{a_h\, h}$$

where

a_h = perimeter of fuel element, ft

h = heat transfer coefficient, BTU/hr-ft^2-°F

3. Fuel centerline temperature

$$t_{CL}(\ell) = t_s(\ell) + q'''a^2 \left[\frac{1}{4K_f} + \frac{\ell n\ b/a}{2K_c} \right]$$

where

a = radius of fuel, ft

b = radius of fuel element, ft

K_f = thermal conductivity of fuel, BTU/hr-ft-°F

K_c = thermal conductivity of clad, BTU/hr-ft-°F

The following correlations were used to determine the various heat transfer coefficient:

1. Boiling convection[18]

$$h = 0.06\ (K/D_e)\ (\rho_f/\rho_g)^{.28}\ (D_e Gx/\mu_f)^{.87} Pr^{.4}$$

where

D_e = equivalent diameter, ft

ρ_f = density of saturated fluid, lb/ft^3

ρ_g = density of saturated vapor, lb/ft^3

G = mass velocity, lb/hr-ft^2

x = quality

μ_f = viscosity of saturated fluid, lb/ft-hr

Pr = Prandtl Number

2. Forced convection[9]

$$h = 0.23\ (K/D_e)\ Re^{.8} Pr^{.4}$$

where Re is the Reynolds Number.

3. Film boiling[12]

$$h = h_{fb} + h_r$$

For film boiling

$$h_{fb} = \left[\frac{h_{fg}\ \rho_f\ \rho_g\ K_g^{\ 3}\ g}{12\ T_{sat}\ L\ \mu_g} \right]^{.25}$$

For radiation[19]

$$h_r = .172\ \frac{\left[(T_1/100)^4 - (T_2/100)^4 \right]\ E}{T_1 - T_2}$$

where

T_1 = absolute temperature of clad, °R

T_2 = absolute temperature of saturated fluid, °R

E = emissivity of clad material

The temperature calculations were performed for a core fully loaded with fuel elements having a power density of 320 watts/liter. This gives the hottest possible core configuration. The mass flow rate used in the calculation was 0.633 lb/sec in each channel. Table 6-1 shows the temperature profile of a single fuel element under normal operating conditions. The maximum temperature is 1518°F, and is the centerline temperature of the fuel element. This shows

Table 6-1

Temperature Profile of a Single Fuel Element
Under Normal Operating Conditions, °F

Length	Coolant Temperatures	Clad Temperatures	Fuel-Clad	Fuel 1/2 Radius	Fuel Centerline
0.00	250.00	280.96	291.78	1211.71	1518.35
0.50	250.00	280.96	291.78	1211.71	1518.35
1.00	250.00	280.96	291.78	1211.71	1518.35
1.50	250.00	280.96	291.78	1211.71	1518.35
2.00	250.00	280.96	291.78	1211.71	1518.35
2.50	250.00	280.96	291.78	1211.71	1518.35
3.00	250.00	280.96	291.78	1211.71	1518.35
3.50	250.00	280.96	291.78	1211.71	1518.35
4.00	250.00	279.36	290.18	1210.11	1516.75
4.50	250.00	276.50	287.32	1207.25	1513.89
5.00	250.00	274.18	285.00	1204.93	1511.57
5.50	250.00	272.25	283.08	1203.00	1509.65
6.00	250.00	270.63	281.46	1201.38	1508.02
6.50	250.00	269.24	280.07	1199.99	1506.64
7.00	250.00	268.04	278.87	1198.79	1505.43
7.50	250.00	266.99	277.82	1197.74	1504.38
8.00	250.00	266.06	276.89	1196.81	1503.46
8.50	250.00	265.24	276.06	1195.99	1502.63
9.00	250.00	264.50	275.32	1195.25	1501.89
9.50	250.00	263.83	274.66	1194.58	1501.23
10.00	250.00	263.23	274.05	1193.98	1500.62

that the fuel element is not molten at any time under normal operation. The maximum clad temperature is 280°F.

A perturbation in the system such as an inadvertant closing of the feedwater inlet valve, would result in a loss of coolant flow. Under this condition the water in the boiler would boil and the two-phase mixture would flow up through the channels by natural convection. It is postulated that under this condition, film boiling would occur in the boiler. The temperature profile for a single fuel element under this condition is shown in Table 6-2. It is seen that the centerline of the fuel will reach a temperature of 1906°F and will be molten. The maximum temperature at one-half the fuel radius reaches a maximum of 1600°F, which is within the allowable limit of the molten radius. Also, the clad temperature reaches 669°F which does not exceed the upper limit of 800°F which will cause excessive clad corrosion. Thus, excessive fuel temperatures will not be a problem, as long as the fuel is covered by water. The NWB was designed to minimize the possibility of leakage by having all pressure vessel penetrations above the fuel, except the downcomer, and by surrounding the boiler with a trench to collect any coolant leakage.

When the fuel is in the storage pool, it must be cooled by forced convection. The temperature profile for a single element under these conditions is shown in Table 5-3. The inlet coolant temperature was assumed to be 80°F and the mass flow was the same as the mass flow through the boiler, 385, 155 lb/hr or 770 GPM. It is seen from the Table 6-3 that all temperatures are well below limits and do not present any problems.

Table 6-2

Temperature Profile of a Single Fuel Element Under Conditions of Film Boiling Along the Entire Channel, °F

Length	Coolant Temperatures	Clad Temperatures	Fuel-Clad	Fuel 1/2 Radius	Fuel Centerline
0.00	250.00	315.05	325.87	1245.80	1552.44
0.50	250.00	463.94	474.77	1394.69	1701.33
1.00	250.00	506.99	517.81	1437.74	1744.38
1.50	250.00	534.16	544.99	1464.91	1771.56
2.00	250.00	554.13	564.96	1484.88	1791.52
2.50	250.00	569.92	580.74	1500.67	1807.31
3.00	250.00	582.95	593.78	1513.71	1820.35
3.50	250.00	594.04	604.87	1524.80	1831.44
4.00	250.00	603.68	614.51	1534.43	1841.07
4.50	250.00	612.19	623.02	1542.94	1849.58
5.00	250.00	619.80	630.63	1550.55	1857.20
5.50	250.00	626.68	637.50	1557.43	1864.07
6.00	250.00	632.95	643.77	1563.70	1870.34
6.50	250.00	638.70	649.52	1569.45	1876.09
7.00	250.00	644.01	654.84	1574.76	1881.40
7.50	250.00	648.94	659.77	1579.69	1886.33
8.00	250.00	653.54	664.36	1584.29	1890.93
8.50	250.00	657.84	668.67	1588.59	1895.24
9.00	250.00	661.89	672.71	1592.64	1899.28
9.50	250.00	665.70	676.52	1596.45	1903.09
10.00	250.00	669.30	680.13	1600.05	1906.70

Table 6-3

Temperature Profile of a Single Fuel Element
Under Pool Storage Conditions, °F

Length	Coolant Temperatures	Clad Temperatures	Fuel-Clad	Fuel 1/2 Radius	Fuel Centerline
0.00	80.00	101.18	112.01	1031.94	1338.58
0.50	82.37	103.56	114.38	1034.31	1340.95
1.00	84.75	105.93	116.76	1036.68	1343.33
1.50	87.12	108.31	119.13	1039.06	1345.70
2.00	89.50	110.68	121.51	1041.43	1348.07
2.50	91.87	113.05	123.88	1043.81	1350.45
3.00	94.25	115.43	126.25	1046.18	1352.82
3.50	96.62	117.80	128.63	1048.55	1355.20
4.00	98.99	120.18	131.00	1050.93	1357.57
4.50	101.37	122.55	133.38	1053.30	1359.95
5.00	103.76	124.93	135.75	1055.68	1362.32
5.50	106.12	127.30	138.13	1058.05	1364.69
6.00	108.49	129.67	140.50	1060.43	1367.07
6.50	110.87	132.05	142.87	1062.80	1371.82
7.00	113.24	134.42	145.25	1065.17	1371.82
7.50	115.61	136.80	147.62	1067.55	1374.19
8.00	117.99	139.17	150.00	1069.92	1376.56
8.50	120.36	141.55	152.37	1072.30	1378.94
9.00	122.74	143.92	154.75	1074.67	1381.31
9.50	125.11	146.29	157.12	1077.05	1383.69
10.00	127.48	148.67	159.49	1079.42	1386.06

B. Critical Heat Flux

There are many correlations in the literature for predicting the critical heat flux. One correlation for the critical heat flux for a saturated pool boiling condition was developed by Rohsenow and Griffith[20]. Since the fluid in the NWB is saturated and at a low flow rate, it closely approaches the saturated pool boiling condition. This correlation was used and is given by

$$q''_c = 143\ h_{fg}\ \rho_g \left[\frac{\rho_f - \rho_g}{\rho_g}\right]^{0.6}$$

where

h_{fg} = latent heat vaporization, BTU/lbm-°F

ρ_g = density of saturated vapor, lb/ft^3

ρ_f = density of saturated fluid, lb/ft^3

From this correlation the critical heat flux was calculated to be 5.5×10^5 BTU/hr-ft^2. The maximum heat flux produced by a 320 w/l fuel element is 1.4×10^3 BTU/hr-ft^2. Comparing these values indicates that the heat flux of the fuel element is not sufficient to cause departure from nucleate boiling. Nevertheless, a film boiling condition was assumed to be possible and the temperature profile calculated is shown in Table 6-2. These results indicate that fuel element meltdown is not a significant problem.

C. Heat Removal System

1. Primary HRS

Each boiler section will be equipped with a primary heat removal system as well as an emergency system. The primary system

is manually operated and is used for normal refueling procedures or to replace the emergency system which is automatically activated. The HRS system normally takes suction from the storage pool and injects the water into the base of the boiler where it flows upward through the core. The water exits through the top of the boiler and is discharged into the storage pool. This system, equipped with two 2000 GPM pumps, is also capable of taking suction from the pipe trench sump or from an auxiliary source. One pump is sufficient to remove the heat from all the fuel elements in the system. The pool storage tank is kept at a suitable temperature by circulation through the air conditioning cooling towers. If the entire fuel element load is placed in the storage pool, the pool cooling will be routed through the heat exchanger to add an additional barrier against leakage of radioactivity.

2. Emergency HRS

The emergency heat removal system operates similarly to the primary heat removal system, except that this system is equipped with motorized valves which are activated in case of loss of flow or when an excess radiation level is detected in the steam supply system. When activated, valves 1-S-2 and 1-C-2 and 1-C-3 will close, and valves 1-E-1 and 1-E-2 will open. See Figures 4-3 and 4-4. One of the system pumps will also be energized to aid in the emergency. If prolonged use is required, this system will be replaced by the primary heat removal system which requires manual activation.

The valves indicated above would be actuated for a loss of flow in Section I. The equivalent valves in each section would be activated for a problem in that section.

D. Radiation Monitoring

Air monitors will be installed at various locations both within the boiler plant and outside. Additionally, portable survey instruments will be available for routine survey throughout the facility.

In the Nuclear Waste Boiler, the steam is generated in direct contact with the nuclear waste fuel element. The primary coolant is circulated to the mechanical equipment as steam, and is returned to the boiler as condensate. Therefore, precautions must be taken to insure that fission products are not distributed through the primary system. Fission products can only be released through a failure of one or more of the fuel element containers. Even if the container fails, the radioactive material in the container is suspended in a monolithic solid with a relatively low solubility in water. Therefore, the probability of fission product contamination of the steam supply system is low.

However, as a safety precaution, fission product monitors will be located in the steam drum of each boiler section. If fission products are detected, the emergency heat removal system will be activated. This will limit the fission product release to the boiler and the storage pool, and will remove the fission products through routine filtration of the storage pool. The problem of determining the source of the radioactive material will still remain.

Removal of the boiler cover and a survey of the fuel elements in the section will be necessary to locate the failed elements.

VII. ECONOMIC ANALYSIS

It has been shown that a Nuclear Waste Boiler can supply the low pressure steam required to provide the heating and cooling requirements for approximately 750,000 square feet of office space. However, for the NWB to be of value, it must be economically competitive with a fossil fueled heat source in providing this required steam. Therefore, to determine any economic advantages, the significant capital and recurring costs for both alternatives must be identified and compared. The base date for this comparison was chosen to be the placed into service date. All costs were based on current cost data and were escalated as discussed later. It was assumed that all air conditioning equipment would be common to both systems and was not included in the analysis. The analysis was performed for an economic life of 15 years and for both government and private financing at a 10% interest rate.

A. Capital Costs

A detailed engineering cost estimate was made for both the NWB design previously presented and for a fossil fueled steam generation system which would provide the equivalent steam capacity. The fossil fueled boiler selected for this comparison was the Trane, Fire Tube Boiler, Model 6B, 600 hp, with a capacity of 20,700 lb/hr. Two boilers will be required to supply the maximum steam demand of 36,000 lb/hr. Having two boilers will provide a backup capability. If one boiler is down for any reason, the remaining boiler will have sufficient capacity to carry the entire heating load or 58% of the

cooling load. The detailed estimates are shown in Table 7-1 and are summarized as follows:

Nuclear Waste Boiler	$610,000
Fossil Fueled Boiler	$107,000

B. Operating Costs

The operating costs in this comparison refer to the labor costs associated with the physical operation of each alternative. Each plant will require continuous manning with three eight-hour shifts per day. At 40 hours a week per shift, four crews will provide for 160 hours of the 168-hour week. Therefore, a fifth crew is required to provide continuous manning. This crew also provides the fill-in capability for sick and annual leave, and an additional maintenance capability.

The fossil fueled plant will have a crew of one boiler operator per shift. Additionally, an operator/mechanic will be required on a 40-hour week basis to perform the maintenance of instruments and controls and for the testing and treatment of the boiler feedwater. The wage rate for the boiler operator is $4.00 an hour, and $5.00 an hour for the operator/mechanic[24]. Thus, a total of six men will be required to operate the fossil plant.

For the NWB it was determined that two operators will be needed per shift. As with the fossil plant, an additional operator/mechanic will be working on a 40-hour week to perform maintenance and testing. He would also assist in the refueling process. It was assumed that the wages for the nuclear plant personnel will be approximately 25%

Table 7-1

Capital Cost Estimates[21,22,23]

Item	Quantity	Unit Cost	Subtotal	Total
I. Nuclear Waste Boiler Plant				
1. Building	6160SF	3.25/SF	20,020	20,020
2. Excavation	3730CY	1.15/CY	4,290	4,290
3. Compaction	150CY	2.50/CY	375	375
4. Concrete (including forms & reinforcing)				
a. Boiler				
(1) Foundation	53CY	45/CY	2,385	
(2) Walls	300CY	105/CY	31,500	
(3) Heads	17.4CY	47/CY	818	
b. Pipe Trench Walls	300CY	90/CY	27,000	
c. Refueling Pool	95CY	90/CY	8,550	
d. Storage Pool				
(1) Foundation	55CY	45/CY	2,475	
(2) Walls	316CY	90/CY	28,440	
e. Slab	114CY	41/CY	4,674	105,842
5. Stainless Steel Liner ($40ft^2$/SH)	60SH	105/SH	6,300	6,300
6. Epoxy Liner				
a. Refueling Pool	3200SF	1/SF	3,200	
b. Storage Pool	3200SF	1/SF	3,200	6,400
7. Piping				
a. 12"	40LF	37/LF	1,480	
b. 6"	300LF	11/LF	3,300	4,780
8. Minor Equipment				
a. Motorized Gate Valves				
(1) 12"	4EA	1,600/EA	6,400	
(2) 6"	12EA	800/EA	9,600	
b. Manual Gate Valves				
(1) 12"	4EA	1,105/EA	4,420	
(2) 6"	24EA	400/EA	9,600	
c. Flow Meter	10EA	200/EA	2,000	
d. Miscellaneous Fittings			5,000	37,020
9. Major Equipment				
a. Overhead Cranes				
(1) 35-ton	1EA	30,000/EA	30,000	
(2) 1-ton	1EA	8,000/EA	8,000	
b. Water Storage Tank (100,000 gal.)	1EA	26,000/EA	26,000	
c. Heat Exchanger (35×10^6 BTU/hr)	2EA	20,000/EA	40,000	
d. Pumps				
(1) Heat Removal	2EA	3,500/EA	7,000	
(2) Sump	2EA	3,500/EA	7,000	
(3) Pool	2EA	3,500/EA	7,000	

Table 7-1 (cont.)

Item	Quantity	Unit Cost	Subtotal	Total
e. Steam Drums	4EA	3,000/EA	12,000	
f. Emergency Generator (150KW)	1EA	18,000/EA	18,000	155,000
10. Radiation Monitoring Equipment				
a. Air Monitor	6EA	600/EA	3,600	
b. Fission Product Monitor	6EA	2,000/EA	12,000	
c. Portable Survey	4EA	400/EA	1,600	17,200
11. Plumbing	6130SF	.45/SF	2,760	2,760
12. Electrical				
a. Lighting and Recepticles	6130SF	.5/SF	3,065	
b. Control Center	job		10,000	
c. Control Wiring	job		10,000	23,065
13. Heating and Ventilation	6130SF	.35/SF	2,150	2,150
14. Fuel Element Support	13,200LB	1.35/LB	17,847	17,847
	Total			403,049
	25% Overhead and Profit			100,762
				503,811
	10% Contingency			50,381
				554,192
	10% A and E			55,419
				609,611
	Say $610,000			

II. Fossil Fueled Boiler Plant

Item	Quantity	Unit Cost	Subtotal	Total
1. Site Preparation	105CY	7/CY	735	735
2. Slab on Grade	53CY	41/CY	2,132	2,132
3. Building (pre-engineered steel 40' x 70')	2800SF	3/SF	8,400	8,400
4. Electrical	2800SF	.5/SF	1,400	1,400
5. Plumbing	2800SF	.45/SF	1,260	1,260
6. Heating and Ventilation	2800SF	.35/SF	980	980
7. Fuel Oil Storage Tank (30,000 gal.)	2EA	8,000/EA	16,000	16,000
8. Water Storage Tank	1EA	8,000/EA	8,000	8,000
	Total			38,907
	25% Overhead and Profit			9,727
				48,634
	10% Contingency			4,863
				53,497
	10% A and E			5,350
				58,837
Boiler Trane Model 6B 600 HP, 20,700 lb/hr 20,085 MBH	2EA	24,000/EA	48,000	48,000
	Total Estimate			106,847
	Say $107,000			

higher. Therefore, the wage rates used were $5.00 per hour for the operator, and $6.00 per hour for the operator/mechanic. A total of 11 men would be required to operate the nuclear plant.

All annual labor costs were increased by a 25% overhead factor to allow for overtime and leave, and were escalated at a 3% per year annual rate to cover increases in the cost of living. The annual operating costs are summarized as follows:

Nuclear Waste Boiler	$145,600
Fossil Fueled Boiler	$ 65,000

C. Maintenance Costs

Annual maintenance costs for the fossil fueled boiler include the minor repairs of the fuel oil pumps, feedwater pumps, fuel service lines, fuel tanks and other related equipment. It was estimated that these costs would be $2,000 a year.

The NWB system contains a large number of safety components which require frequent testing to insure their reliability. These components include motor operated valves, pumps, emergency generators, heat exchangers and water storage and transfer pumps. A continuous testing and preventative maintenance program will be necessary, and the annual cost for this program was estimated to be $5,000. These annual maintenance costs were also increased at an annual rate of 3%. The annual maintenance costs are summarized as follows :

Nuclear Waste Boiler	$5,000
Fossil Fueled Boiler	$2,000

D. Fuel Costs

The cost for the fuel oil for the fossil fueled boiler was determined from the normalized steam demand curve developed earlier. The current price for No. 2 fuel oil is 21 cents per gallon, or \$8.82 per barrel[24].

The heat content of 19,460 BTU/lb was used for the fuel oil in developing the cost factor. Assuming an 80% combustion efficiency, the cost of steam was calculated to be \$0.001423 per lb. The normalized steam demand curve as developed for 186 tons of air conditioning, has a normalization factor of 1.577×10^6 lb/month. Adjusting this to 2,500 ton of air conditioning, the maximum monthly steam requirement was 1.696×10^7 lb/month. Multiply the steam demand factor by the cost factor, the normalized steam cost factor was determined to be \$24,134 per month. Based on the heating and air conditioning requirements, 704,000 gallons of fuel oil at a cost of \$148,765 are required annually.

Coal was considered as a fuel, but was rejected because of the relatively small size of the system. The dominant consideration in using coal is that it is a much less attractive fuel than oil or gas, due to its solid state; and due to the problems created from its higher waste content at every stage, from mining to post-combustion waste disposal. Coal will usually be consumed only when its costs are sufficiently below those of the other fuels to outweigh these disadvantages[25].

Since there is no present use for nuclear waste, a price has not been established. Therefore, it has been assumed that the

transportation costs to the NWB are less than the cost of interim storage, and that the fuel is provided at no cost to the user. However, in the economic analysis, the cost of the nuclear waste in \$/Kg was determined to show the maximum price that can be paid by the user for the NWB fuel, including transportation, and still compete with a fossil fueled boiler. Since each boiler section contains 42,633 Kg of waste, the maximum price can be determined by dividing the annual saving by 42,633 Kg.

The price of oil is expected to increase considerably over the next 20 years. During 1973, local fuel costs have increased from \$.17 to \$.21 per gallon. This represents a 19% increase. This increase is considered unusual, therefore, a 5% annual escalation factor was used in the comparison. The annual fuel costs are summarized as follows :

Nuclear Waste Boiler	\$ 0
Fossil Fueled Boiler	\$148,765

E. Analysis

Several economic analysis techniques are available which are very useful in comparing alternatives. One of these techniques is the Savings/Investment Rate (SIR) method as described in reference 26. In this method the Savings/Investment ratio is determined by dividing the cumulative present value of future savings, discounted at the prescribed interest rate, by the present value of the difference in capital investment between alternatives. For example, assume Concepts A and B both satisfy the design requirement, and A requires

a greater initial investment but has a lower annual expense. The present value of the annual savings of Concept A are then divided by the present value of the difference in capital investment. If the ratio is greater than one, the choice of Concept A over Concept B is justified, in that the additional initial investment will be recovered in savings over the economic life of the plant.

The cumulative present value of the annual operating costs (CAOC) is given by

$$CAOC = \sum_t AOC(t)\ PWF\ (x,t).$$

AOC(t) is the annual operating cost and PWF(x,t) is the present worth factor given by

$$PWF(x,t) = (1-x)^{-t}.$$

In this relationship x is the prescribed interest rate, and t is the economic life of the plant in years. The estimated annual costs for each alternative are summarized below:

	Fossil Fueled Boiler	Nuclear Waste Boiler
Operating Costs	\$ 65,000	\$145,600
Maintenance Costs	2,000	5,000
Fuel Costs	148,765	0
Total	\$215,765	\$150,600

As discussed earlier, the nuclear fuel was assumed to be provided at no cost, and further analysis will be performed to determine the maximum cost that can be paid for the fuel. The annual savings in operating costs are \$65,165 for the first year. These annual costs were escalated as described earlier and were discounted at 10%. The cumulative savings in annual costs over the 15-year economic life of the plant are \$797,959.

To calculate the present value of the capital costs, it was assumed that the construction time for each plant would be one year, and that the partial construction cost payments would be made in 12 equal installments. The present value of the capital costs is given by

$$CC = \sum_{t} ICC(t)\ CAF(x,t).$$

ICC(t) is the partial construction cost payment, CC/12, and CAF(x,t) is the compound amount factor at 1/12 the annual interest rate for 12 periods. The compound amount factor is given by $CAF = (1+x)^t$, where x is the annual interest rate divided by 12 and t is 12 periods. The capital costs and the present worth of the costs are summarized below:

	Fossil Fueled Boiler	Nuclear Waste Boiler
Capital Cost	\$107,000	\$610,000
Present Worth	112,043	638,749

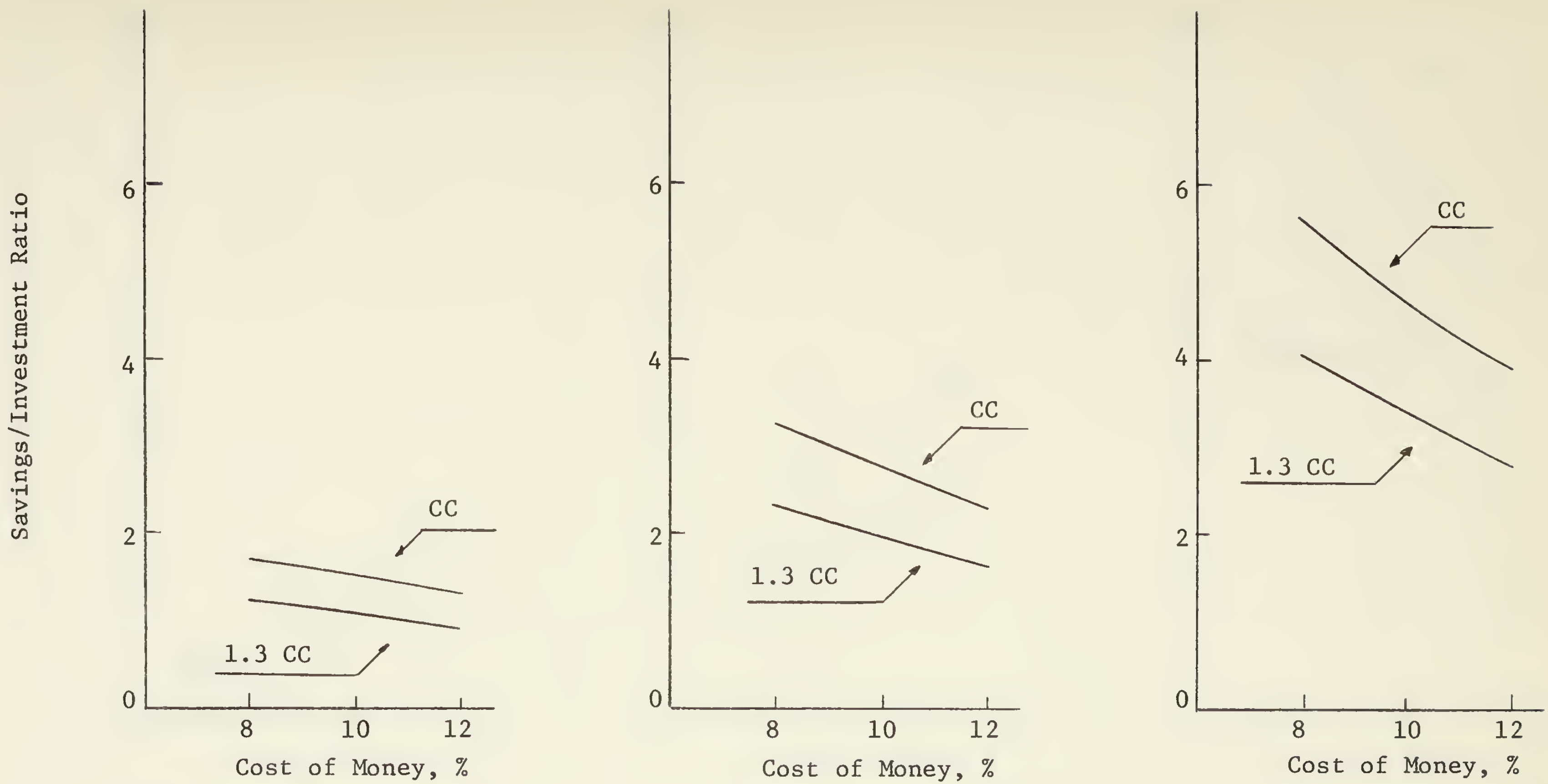
The difference in the present worth of the capital costs is \$526,706 and the Savings/Investment ratio is given by

$$SIR = \frac{\$797,959}{\$526,706} = 1.514$$

Based on the SIR, the NWB presents an economic advantage over the fossil fueled boiler.

In this economic analysis, there is a degree of uncertainty in the original capital investment of the NWB, the fuel oil cost escalation, the cost of future money, and the economic life of the alternatives. The sensitivity of these parameters was analyzed by allowing the parameters to vary and determining the SIR at the various values. The initial investment for the NWB was increased by 30% to \$800,000 and the SIR was determined at 8%, 10% and 12% interest rates over economic lifetimes of 15, 20 and 25 years with fuel oil escalations of 5%, 10% and 15%. The results are shown in Figures 7-1, 7-2, and 7-3.

The results of these comparisons were generally as expected. They show that the relative attractiveness of the NWB decreases with the increased cost of money, but increases with the greater escalation of the fossil fuel costs. Additionally, the attractiveness of the NWB increases with the increase in economic life. Under all conditions analyzed, except one, the SIR for the NWB is greater than one, indicating that the choice of the NWB would represent a savings. The exception is in Figure 7-1 for a 30% increase in the NWB investment at a 12% interest, with 5% fuel cost escalation for a 15-year economic life. From the Figures, it is also obvious that the fuel



a. 5% Fuel Escalation

b. 10% Fuel Escalation

c. 15% Fuel Escalation

Figure 7-1

Savings/Investment Ratio for NWB Compared with Fossil Fueled Boiler
Includes Variable Fuel Costs, Money Costs, and NWB Capital Costs
15-Year Economic Life

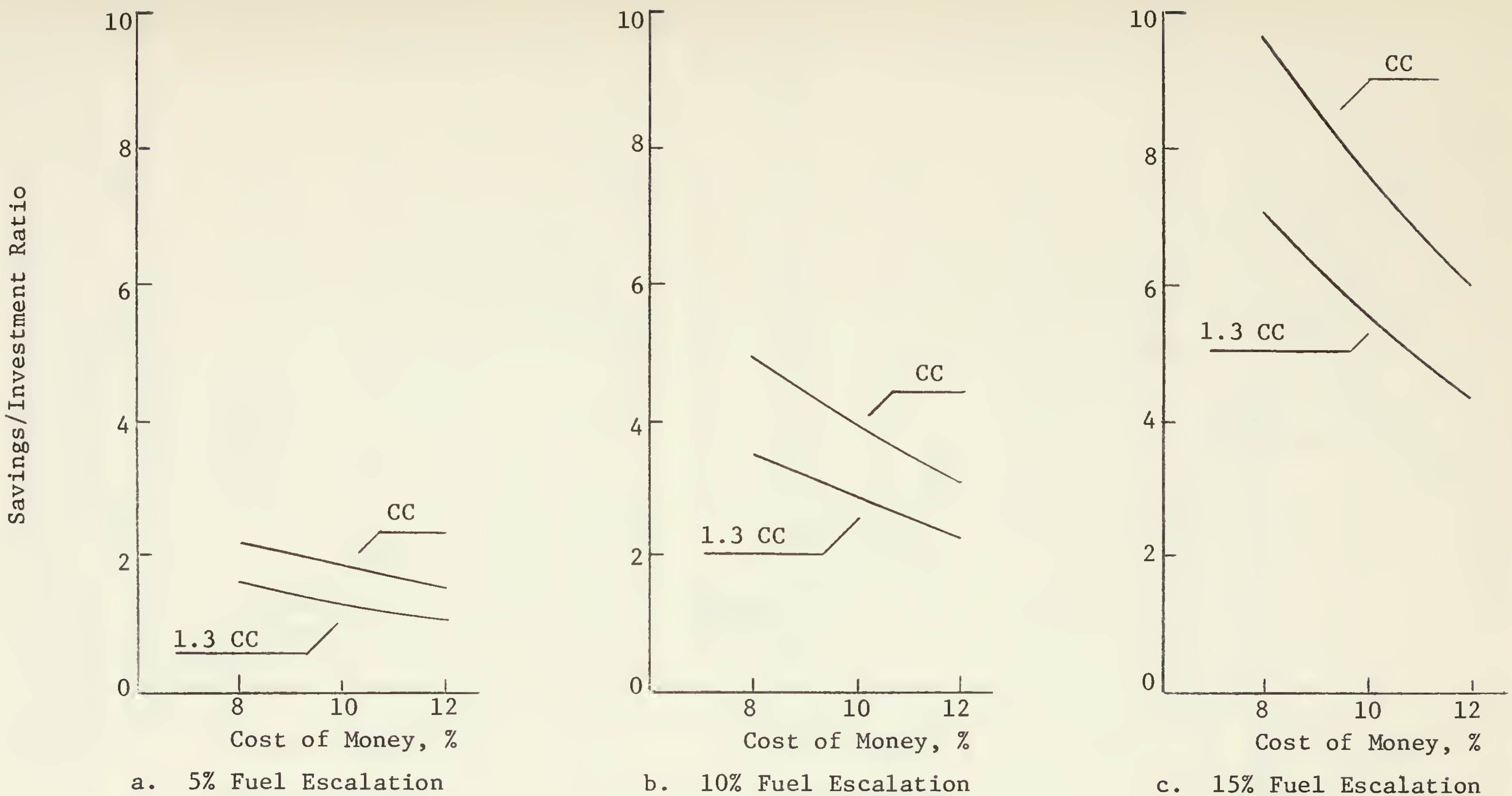
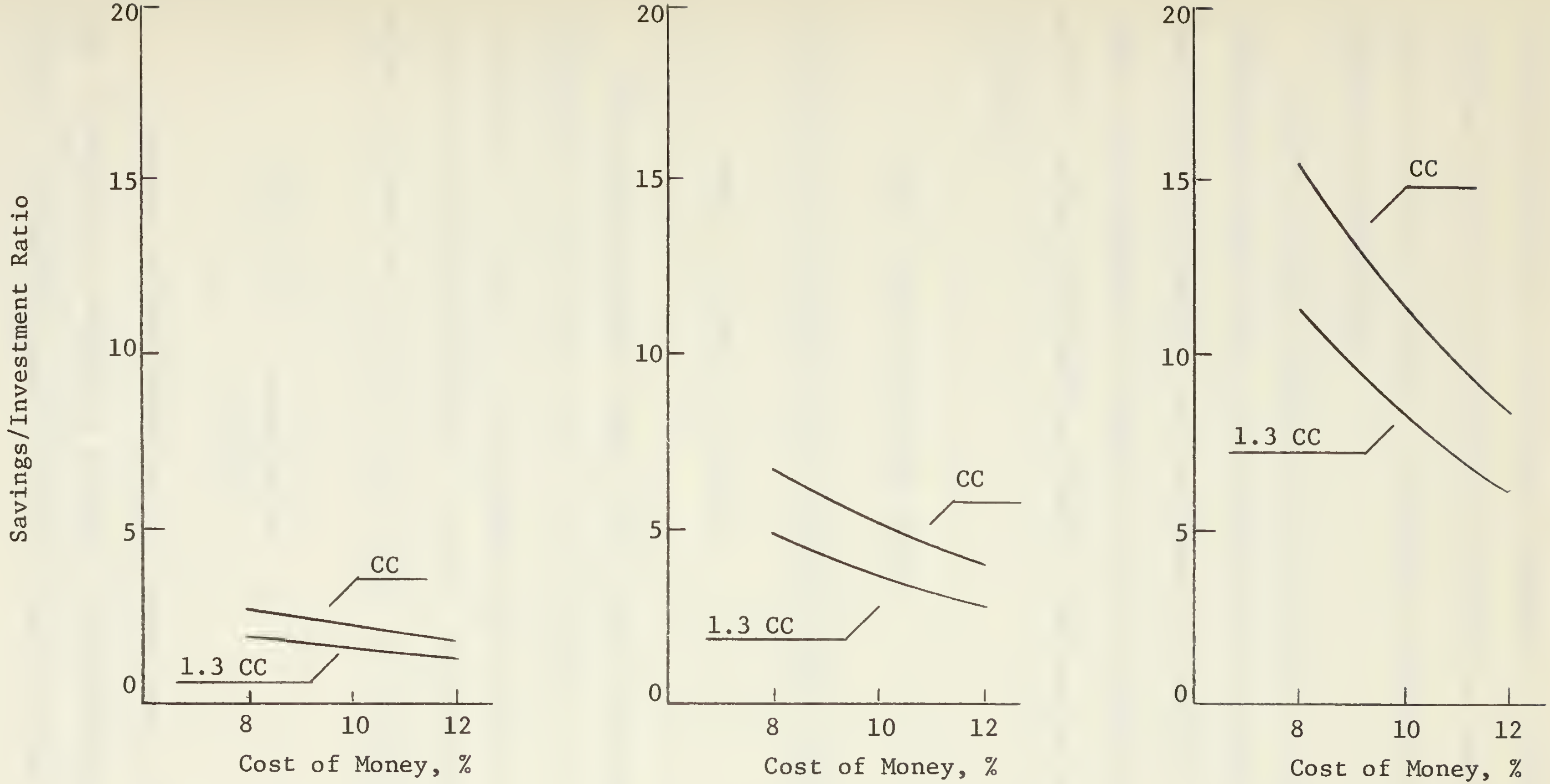


Figure 7-2

Savings/Investment Ratio for NWB Compared with Fossil Fueled Boiler
Includes Variable Fuel Costs, Money Costs, and NWB Capital Costs
20-Year Economic Life



a. 5% Fuel Escalation

b. 10% Fuel Escalation

c. 15% Fuel Escalation

Figure 7-3

Savings/Investment Ratio for NWB Compared with Fossil Fueled Boiler
Includes Variable Fuel Costs, Money Costs, and NWB Capital Costs
25-Year Economic Life

oil cost escalation is the dominant parameter. If the oil costs continue to rise at high rates, 15%, the NWB can have a SIR greater than 15, and would represent a tremendous economic advantage over the fossil fueled system.

Another economic comparison was performed. In this analysis the average annual cost, or the average annual revenue required to operate each alternative, was determined and compared. The average annual cost includes all operating cost, the cost of capital recovery, and the depreciation and tax effects. The average annual cost, AAC, is given by

$$AAC = AAOC + (CC)\ CRF\ (x,t).$$

The value of the average annual operating costs, AAOC, is given by

$$AAOC = (CAOC)\ CRF'\ (x,t)$$

The capital recovery factor, CRF' (x,t), is the number that when multiplied by the present worth of some quantity, such as capital investment, will result in the average annual revenue required to pay off the capital investment. The CRF' (x,t) assumes no salvage value, and does not include income taxes. It is given by

$$CRF'\ (x,t) = \frac{x}{1-(1-x)^{-t}}$$

To account for the added cost of income taxes, the CRF' (x,t) must be modified. Assuming a straight line depreciation for tax purposes, the modified capital recovery factor becomes

$$CRF\ (x,t) = \frac{CRF'\ (x,t) - r/t}{1-r}$$

where r is the effective income tax rate[27]. The application of the CRF (x,t) to the present value of the capital costs effectively averages this cost over the depreciation time of the plant.

The application of the CRF' (x,t) to the cumulative present value of the annual operating costs average these costs over the economic life of the plant. The taxes are not included in the CRF' (x,t), since operating cost expenses are not taxable. This technique was developed by Girvin and Witzig[28], Dade[27], and Zodiaco[29].

In the case of government financing, the income tax rate was equal to zero, and the CRF (x,t) becomes CRF' (x,t). This technique must be slightly modified when private financing is concerned. In this case other factors such as equity return rate, income tax rate, and debt to capital ratio must be included. These factors effectively reduce the interest rate. Thus, the effective cost of money is given by[27]

$$x = jb + i(1-b) - jbr$$

where

j = interest rate = 9%

b = debt to capital rate = 0.071

i = equity return rate = 3.8%

r = corporate tax rate = 57.3%

for a large United States chemical corporation[15]. Thus, the effective cost of money is 3.58%.

This average annual cost technique was utilized for both governmental and private financing. The results are summarized in Tables 7-2 and 7-3. Additionally, these tables show the maximum allowable cost for solid waste that can be paid by the user and still be competitive with the fossil fueled system. As discussed earlier, this is determined by dividing the annual savings by the weight of fuel replaced annually.

The allowable cost of the nuclear fuel was further investigated to determine its variation with the cost of fossil fuel and the economic life of the NWB. The results are shown in Figure 7-4. From this figure it is seen that the allowable cost of the nuclear fuel increases with both the increase in the cost of the fossil fuel and the increase in the economic life of the NWB. The \$6/Kg line in the figure is the estimated cost of shipping spent fuel elements from a light water reactor[5], and represents the upper limit of the transportation cost for the NWB fuel. This line also represents an annual savings of \$255,000 for the NWB if the waste is provided at no cost.

All economic analyses presented here indicate that savings can be realized by utilizing the NWB as a heat source in generating low pressure steam required for heating and air conditioning. The magnitude of the savings depends upon the future cost of fossil fuel and the policy toward the acceptance of the NWB as an interim waste storage facility.

Table 7-2

Economic Analysis, 10% Financing
Not Including Tax and Depreciation Effects

Plant Life (years)	Average Annual Cost (dollars) Fossil	Average Annual Cost (dollars) Nuclear	Average Annual Savings Nuclear Over Fossil (dollars)	Maximum Allowable Cost to User for Solid Waste Including Transportation ($/Kg)
15	302,238	266,713	35,525	.83
20	320,208	265,062	55,146	1.29
25	336,665	267,399	69,266	1.62

Table 7-3

Economic Analysis, Private Financing
Effective Cost of Money 3.8%

Plant Life (years)	Average Annual Cost (dollars)		Average Annual Savings Nuclear Over Fossil (dollars)	Maximum Allowable Cost to User for Solid Waste Including Transportation ($/Kg)
	Fossil	Nuclear		
15	313,839	261,739	52,100	1.22
20	347,493	287,912	59,581	1.40
25	383,009	309,063	73,946	1.73

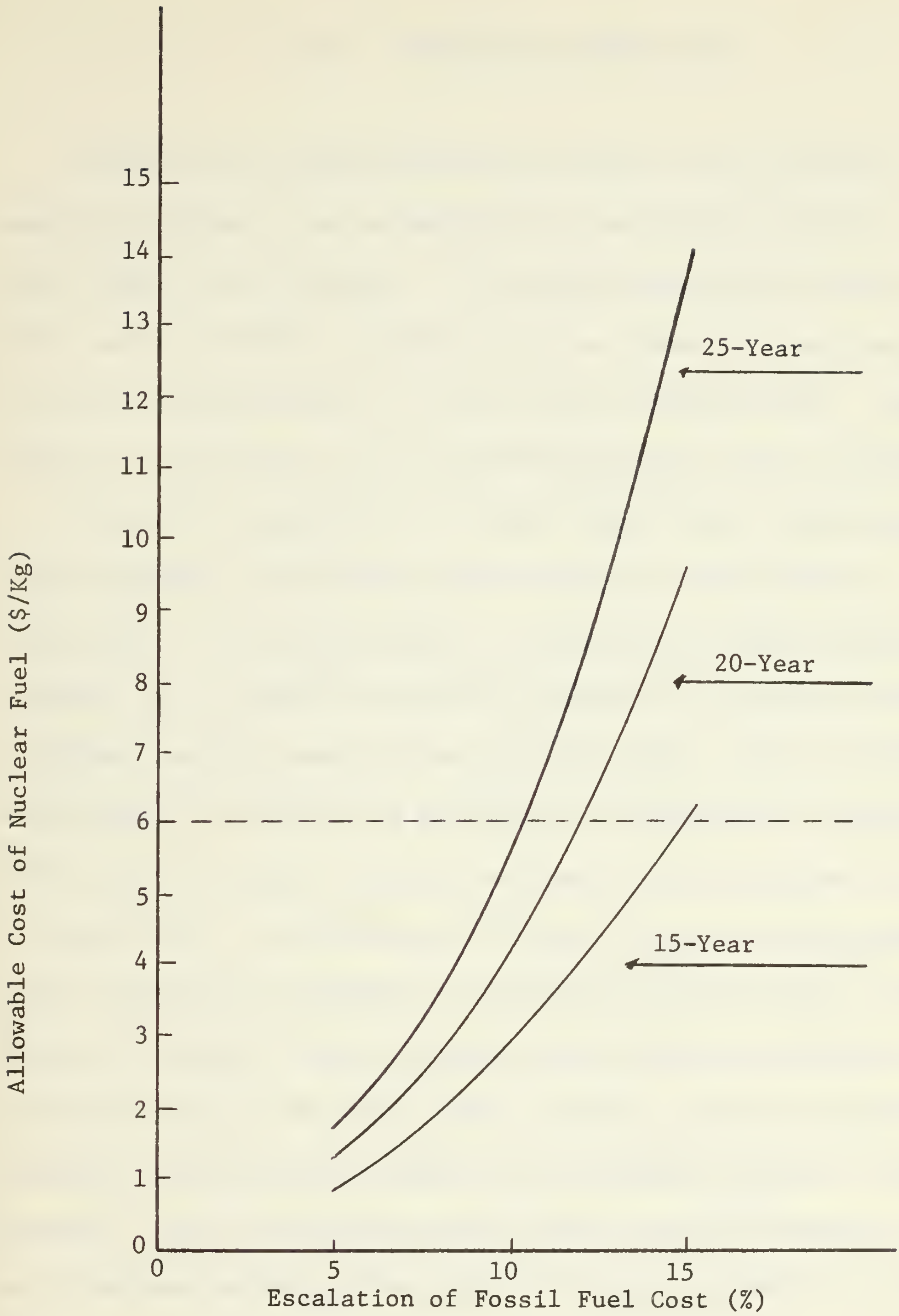


Figure 7-4

Variation of Allowable Nuclear Fuel Cost With Escalation of Fossil Fuel Cost and Economic Life of NWB. 10% Financing.

VIII. SUMMARY AND CONCLUSIONS

As the role of nuclear energy increases in the electrical power generation industry, the accumulation of solidified high-level radioactive waste also increases. It has been proposed that these solidified wastes can be utilized to generate the low pressure steam required to provide a cooling capacity of 2,500 tons of refrigeration. This capacity is sufficient for a 750,000 square foot office building.

The steam is generated in a Nuclear Waste Boiler using stainless steel containers filled with nuclear waste as fuel elements. Analytically, it has been shown that sufficient driving pressure is available in the natural circulating generator to maintain stability throughout the four years in the core fuel cycle. The relatively low power density of the fuel is sufficient to generate the required steam, but is also low enough that fuel meltdown is not a problem in the case of loss of flow. In the event of a loss of flow, two independent heat removal systems are available to provide forced circulation. The system is shielded to keep the dose rates within the building below acceptable limits. The fuel element container and the mechanical equipment provide barriers against the accidental spread of radioactive materials. Only the chilled and hot water from the mechanical units are circulated outside of the plant, and these systems are isolated from the primary system.

The economic benefit of the Nuclear Waste Boiler, when compared with an equivalent fossil fueled system, has been discussed in detail. Savings greater than $50,000 a year over a 20-year economic life for

the NWB are predicted. This assumed a fossil fuel cost escalation of 5% per year. Economic analysis performed to determine the effects of the escalation of fossil fuel costs indicate that greater savings result from the use of the NWB as fossil fuel costs increase. The development of the NWB as an energy source is very attractive in light of today's energy crisis and fossil fuel shortage.

It is concluded that the objectives of this work have been accomplished. The NWB can safely and economically provide the steam required to heat and cool a large office building. The NWB can also provide interim storage for these high-level wastes during a period of relatively high power density in the waste container. In this application, the NWB saves 16,600 barrels of heating oil annually. Further development of the Nuclear Waste Boiler as a part of the overall energy program could result in substantial savings in fossil fuel.

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